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A Compendium of Computer Codes for Light Water Reactor Analysis

Peter P. Bieniarz, Peter G. Prassinos
Energy Incorporated

Dennis Engi
Sandia Laboratories

Prepared by
Sandia National Laboratories
Albuquerque, New Mexico 87185 and Livermore, California 94550
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A COMPENDIUM OF COMPUTER CODES FOR LIGHT WATER REACTOR ANALYSIS

Peter P. Bieniarz
Peter G. Prassinos
Energy Incorporated

Dennis Engi
Sandia National Laboratories

Date Published: December 1981

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Albuquerque, New Mexico 87185
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for the
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ABSTRACT

A compilation of computer codes that can be used to analyze the effects of postulated accidents and failure phenomena upon light water reactor systems is presented. From a safeguards perspective, these codes can be used in the determination of the consequences of acts of sabotage, and therefore, they play a key role in ranking vital areas. Each code is described in terms of its function, its limitations, its use, and a comparison of the output from the code with experimental observations. A list of applicable reference documents related to each code is also presented. Primary emphasis is placed on nuclear power plant loss-of-coolant accident analyses, nuclear power plant transient analyses, and miscellaneous (nontransient) analyses. A tabulation of available codes and their purpose facilitates use of the report in identifying codes applicable to a particular problem.

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SUMMARY

Computer codes represent a powerful analytical tool for assessing the safety of nuclear reactors and their associated systems under various operating conditions. Among these computer codes are a number of codes used for accident and safety analyses of light water reactor (LWR) plants. This report presents a compilation of codes which may be used to study the effects on LWR systems caused by postulated accidents or failure phenomena resulting from premeditated acts of sabotage. Each code description addresses (1) the function accomplished by the code, (2) the code limitations, (3) code utilization and comparisons with experimental observations, and (4) a list of reference documents. The report concentrates on loss-of-coolant accident (LOCA) analyses, transient analyses, and miscellaneous (nontransient) analyses.

The LOCA analysis codes calculate and predict the behavior of an LWR during a postulated accident involving loss of fluid from the primary coolant system. The categories considered include large-break and small-break LOCAs.

The transient analysis codes presented calculate and predict the behavior of nuclear reactor systems during transient events that do not involve loss of fluid from the primary coolant system. Such events might include anticipated transients with and without scrams and operational transients, such as loss of primary flow, loss of secondary flow, reactivity initiated transients, and increased load.

The nontransient codes include steady-state calculations, system performance codes, and probabilistic and reliability analysis codes.

Summary sheets are presented for each computer code. These sheets include sections which describe (1) the purpose of the code, (2) the status of the code (development, availability, and utilization), (3) the problems solved by the code, (4) the systems and subsystems simulated, (5) the modeling approach used, (6) the code limitations, (7) information about the computers upon which the code is operational, (8) future efforts for further code development, (9) the interface/overlap of the code with other codes, (10) reference documents related to the code, and (11) possible contacts from whom more information about the code can be obtained.

A COMPENDIUM OF COMPUTER CODES FOR LIGHT WATER REACTOR ANALYSIS

1. INTRODUCTION

The safety/safeguards analysis of nuclear reactors and their associated systems requires a high degree of mathematical sophistication. Numerical solution procedures associated with these analyses are generally complex and time-consuming, necessitating the development and use of computer codes. These computer codes are powerful analytical tools that have the advantages of accuracy, speed, and low cost.

The number, complexity, and diversity of computer codes developed to analyze light water reactor (LWR), pressurized water reactor (PWR), and boiling water reactor (BWR) systems under various conditions has increased markedly in recent years. The objective of this report is to compile and briefly describe computer codes that can be used for analyzing the effects of accidents or sabotage of LWR systems. This compilation can be used by the analyst to determine which computer codes are available to accomplish a specific task. The four general categories addressed for each code are

1. Function accomplished by the code,
2. Code limitations,
3. Code utilization and comparisons with experimental observations, and
4. Reference documents.

There are many areas of reactor system analysis and application covered by present-day computer codes. This report concentrates on three basic areas of analysis and application:

1. Nuclear power plant LOCA (loss-of-coolant accident) analysis,
2. Nuclear power plant transient analysis, and
3. Miscellaneous (nontransient) analysis.

The LOCA analysis codes calculate and predict the behavior of a nuclear power plant during a postulated incident involving loss of fluid from the primary coolant system. The LOCA situation can be divided into two categories: large-break and small-break LOCAs. A large-break LOCA occurs if fluid escapes the primary coolant system through a break in the primary coolant piping in which the flow area of the break is comparable in size to an entire primary coolant piping cross section. The resulting transient is rapid and generally lasts, at most, a few minutes.

The small-break LOCA occurs if fluid escapes the primary coolant system through a break in the primary coolant piping in which the flow area of the break is equal to or less than approximately 0.5 square feet. A small-break LOCA can also be caused by breaks in the piping connected to the primary coolant system, such as instrument lines, recirculation lines, pressure relief lines, and reactor vessel penetrations. The resulting transient is slow and generally lasts from several minutes to several hours.

Transient analysis codes calculate and predict the behavior of nuclear reactor systems during transient events that do not involve loss of fluid from the primary coolant system. Transient codes that treat situations resulting from the loss of fluid from the secondary coolant systems are, however, included in this report. The transient situations include anticipated transients with and without scrams (ATWS) and operational transients, such as loss of primary flow, loss of secondary flow, reactivity initiated transients, and increased load.

The miscellaneous codes are nontransient codes, which include steady-state calculations, system performance codes, and probabilistic and reliability analysis codes.

The LOCA and transient codes, presented in Appendices A and B, are described in some detail. The miscellaneous codes, presented in Appendix C, are described in less detail. Codes, such as conduction heat- transfer codes that have been developed for general calculations pertinent to reactor safety system analysis are not described, but these codes are listed in Appendix D. (Only codes that apply specifically to reactor safety and nuclear power systems are included.) Codes that deal with radiation exposure and release as well as codes that perform neutronic calculations are not included in this report. But, codes that calculate coupled neutronic-thermal-hydraulic behavior and reactivity-initiated accidents are included. Codes written specifically for gas or fast reactor analysis are not included in this report. There are, however, fast reactor codes that can be directly applied or modified for thermal reactors, and these codes are included.

The remainder of this report is divided into three sections and five appendices. Section 2 contains a sample summary of the information supplied for each code, and Section 3 contains a table of pertinent information for each code. Section 4 sketches a formalized approach for the systems analyst's use of the codes described herein and indicates where additional effort is needed in order to make this approach workable. Appendices A, B, and C contain the summary descriptions of nuclear plant LOCA codes, nuclear power plant transient codes, and the miscellaneous codes, respectively. A list of computer codes that perform general engineering calculations is provided in Appendix D, and the acronyms most commonly used in the nuclear reactor industry are defined in Appendix E.

2. SAMPLE CODE INFORMATION

The following sample summary illustrates the kind of information provided for each code and the format used. The code summaries are divided into several sections; each section contains information on a specific aspect of the code, such as code purpose, status, or problems solved. Although every attempt was made to render each code summary as complete as possible, information pertaining to each section for each code was not always available.

The information given on each code summary was obtained primarily from code description reports and user manuals. Some information was also obtained from journal articles, facility progress reports, and contacts with authors or code users. The reference section of each summary lists the documents from which the information in a particular summary was derived and from which the reader may obtain a more detailed description of the code.

These code summaries provide the analyst with a prefatory description of the codes used for LWR safety analysis. Once the particular problem requirements are defined, the information contained in the code summaries will allow the analyst to select the appropriate code or codes needed to treat significant failure phenomena based on the applicability, complexity, and cost of the code.

Sample Code Summary
(Information and Format)

Purpose

This section of the code summary contains a brief statement concerning the nature of the physical problem solved and/or the code objective.

Status

A. Development Subsection A contains the name of the development organization and the name of the current developmental version, if it is different from the summary title.

B. Availability Subsection B contains the names of various versions of the code, their documentation or availability status, and the locations or organizations that have run the code. The availability status of a code is classified as one or more of the following:

Developmental -- The code is still in an early stage of its development or is in the process of being debugged. The code has not been made to run successfully on a computer outside of the development facility, but it may be partially operational at the development site.

Operational -- The code is being run successfully on a computer at or outside of the development facility. This implies the code is debugged and ready for use by individuals other than the developer. An operational code is generally accompanied by a user's manual and some documentation.

Released -- A released status for a code implies that it has been fully documented and is available through the National Energy Software Center (NESC), Argonne National Laboratory, 9700 South Cass Avenue, Argonne, Illinois 60439. Restrictions on the use of the code are noted, as is the cost of the code package.

Proprietary -- A proprietary code is not available for use outside of the development facility, except through contractual arrangements with the developer. Terms and conditions are included in the summaries, when available.

C. Utilization Subsection C describes the current use of the code and the specific safety problems to which it has been or is being applied.

Problems Solved

This section contains a listing of the various phenomena or physical problems that are treated.

Systems and Subsystems Simulated

This section lists the specific safety systems as well as other systems that are simulated by the code.

Modeling

This section contains a description of the mathematical models, basic physical approximations, and the numerical solution methodology contained in the code.

Limitations

This section describes the limitations to the use of the code that result from the amount of storage allocated, the use of approximations, and the restrictions on the complexity of the physical problem or phenomena that can be treated.

Computer Operations

This section contains a list of the computers upon which the code has been made operational and the code logistics, including a description of the core memory requirements and special peripherals, the mode of operation (interactive or batch), the programming language and compiler requirements, special systems routines, and special plotting software.

In addition, it gives the computer run time for the solution of a typical problem and a brief description of the problem (geometry, real time, number of nodes, etc.). The section of the code requiring the major consumption of time is identified.

Future Efforts

This section describes anticipated or planned improvements or modifications to the code.

Interface/Overlap

This section lists other codes with which the described code interfaces to provide or receive data and similar or equivalent codes.

References

This section lists the material from which information was derived or from which additional details can be obtained.

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This section contains a description of the mathematical models, basic physical approximations, and the numerical solution methodology contained in the code.

Limitations

This section describes the limitations to the use of the code that result from the amount of storage allocated, the use of approximations, and the restrictions on the complexity of the physical problem or phenomena that can be treated.

Computer Operations

This section contains a list of the computers upon which the code has been made operational and the code logistics, including a description of the core memory requirements and special peripherals, the mode of operation (interactive or batch), the programming language and compiler requirements, special systems routines, and special plotting software.

In addition, it gives the computer run time for the solution of a typical problem and a brief description of the problem (geometry, real time, number of nodes, etc.). The section of the code requiring the major consumption of time is identified.

Future Efforts

This section describes anticipated or planned improvements or modifications to the code.

Interface/Overlap

This section lists other codes with which the described code interfaces to provide or receive data and similar or equivalent codes.

References

This section lists the material from which information was derived or from which additional details can be obtained.

Contacts

The section lists the person(s) (and agency) from whom the code package or information about the code may be obtained.

3. CODE INFORMATION SUMMARY

A summary of the pertinent information for each code is contained in Table 1, which provides a convenient means of identifying the areas of analysis and applicability for which each code has been developed. The data contained in Table 1 include the code name, the development organization, the area of analysis and application of the code, the availability of the code, the computer or computers on which the code is operational or being developed, the typical program run time (when available), and the cost of obtaining the code package, if this is possible.

The code development organization (the company or facility that originally developed the code) is identified in the first column by an acronym. Definitions of these acronyms are given in Appendix B. The year that the code was released or published is indicated in the second column. The specific areas of analysis and application for each code are indicated by entries in the third, fourth, and fifth columns, headed "LOCA Codes," "Other Transient Codes," and "Miscellaneous Codes," respectively. These entries consist of a number (1 or 2) followed by a letter or a series of letters. The number (1) indicates that the code is a total system code; the number (2) indicates that the code treats the behavior of a separate component of a system. The letters indicate the system or component analyzed by the code (i.e., primary loop, containment, etc.) and the type of analysis performed (i.e., thermal, hydraulic, etc.). Definitions of the numbers and letters are given in a legend provided at the right of the table. Columns indicating the "availability" of the code (letter entry defined in legend), the "computer" on which the code has been implemented, the "run time" for the code (in seconds), and the dollar "cost" of the code complete the summary information. An example of Table 1 entries for a specific code is contained in the following paragraph. Table entries are shown in parentheses.

Consider the RETRAN code described on page A-103. This code was developed by the Electric Power Research Institute (EPRI) and was published in 1978 (78). The code is applicable to the total reactor system (1), primarily, to all systems and components contained in the reactor's primary loops (PL), and the code can be used for LOCA analysis as well as the analysis of other transients such as ATWS and operational transients. The code is proprietary (P), and it is operational on both the CDC and IBM computers (CDC/IBM). The typical program run time is about 3,600 seconds on the CDC 6600 computer and about 18,000 seconds on the IBM 360 computer (3600/18000). The cost of obtaining the code is $10,000 (10000).

The analyst can use Table 1 to make an initial assessment of the codes that may be applicable to the problem under consideration. More details on the particular codes listed are found in the code summaries given in Appendices A, B, and C.

Table 1

Summary Information

CODE NAME	DEVELOPMENT ORGANIZATION	YEAR RELEASED OR PUBLISHED	LOCA CODES	OTHER TRANSIENT CODES	MISCELLANEOUS CODES	AVAILABILITY	COMPUTER	RUN TIME (SECONDS)	COST ($)
ADVENT	DPC	72	2CT			R	GE 635	—	67
AIROS2A	AI	68		2C,F		R	IBM 360	120	68
ALLCUTS	HEDL	75			FT	O	CDC CYBER 70	—	—
ALMOD	GRS	77		1PL,S		P	IBM 360	2 TO 4 RS	—
ASCOT-1	JAERI	78	1PL			O	FACOM 230-75	—	—
AXTHRM	LANL	64		2C,F		R	IBM 7094	1 TO 3	—
BEACON/ MOD2A	EG&G	78	2CT			R	CDC 7600	40	138
BEHAVE	GE	72		2F		R,RE	HONEYWELL 6000 CDC 7600	V	70 85
BLOWDN-2	W̄	70	1PL			P	CDC 7600	V	--
BOIL1	BCL	77		2C		R	CDC 7600	40	57
BURST-1	INC	70	2PL			R	IBM 7044	60	61
CADD	B&W	73		2F		P	CDC 6600	—	—
CESEC	CE	74		1PL		P	CDC 7600	—	—
CHART	ORNL	78		2CT		R	IBM 360/370	15	64
CHEMLOC-2	ANL	69	2C,F			R	CDC 3600	120	61
CHIC-KIN	W̄	70		1PL		RE	CDC 6600	—	68
CLAPTRAP	UKAEE	77	2CT			O	—	—	—
COAST	CE	73		1PL		P	CDC 6600	--	--

LEGEND

1 = TOTAL SYSTEM
2 = COMPONENT
C = CORE
CT = CONTAINMENT
D = DEVELOPMENTAL
F = FUEL
FT = FAULT TREE, RELIABILITY, ETC.
H = HYDRAULIC
O = OPERATIONAL
P = PROPRIETARY
PL = PRIMARY LOOP
R = RELEASED
RE = RESTRICTED USE
RS = TIMES REAL TIME
S = SECONDARY
SS = STEADY STATE
T = THERMAL
TS = PER TIME STEP
V = VARIABLE, PROBLEM-DEPENDENT

Table 1 (continued)

Summary Information

CODE NAME	DEVELOPMENT ORGANIZATION	YEAR RELEASED OR PUBLISHED	LOCA CODES	OTHER TRANSIENT CODES	MISCELLANEOUS CODES	AVAILABILITY	COMPUTER	RUN TIME (SECONDS)	COST ($)
COBRA	BNWL	78	2C,F			R	CDC IBM	6 4	72 96
COCO	W̄	74	2CT			P	CDC 7600	—	—
COMCAN	EG&G	76			FT	R	CDC CYBER 76 IBM 360	0.4 76	— 70
COMPARE	LANL	77	1PL			R	CDC IBM	0.002 TS 0.012 TS	61
COMPERCII	CE	74	2C			P	CDC 7600	—	—
CONTEMPT-LT/026	ANC	70	2CT			R,D	IBM 360	0.021 TS	119
CONTEMPT 4/MOD2	EG&G	79	2CT			R,D	CDC 7600	30	—
CONSTRANS	CE	74	2CT			P	CDC 7600	—	—
CRAFT	B&W	74	1PL			P	CDC 6600/ 7600	V	—
DINO	DAEA	72	2F			O	—	—	—
DSNP	ANL	78		1PL		R	IBM 370	—	101
DYNAM	GA	70		2PL		R	UNIVAC 1108	5	66
ECCSA4	BCL	71	2F			R	CDC 6400	200 RS	66
EFLOD	RPI	74	1T,H			D	—	—	—
ELRAFT	W̄	71			FT	O	CDC 6600	72	—
FACTRAN	W̄	72		2F		P	CDC 6600	—	—
FATRAM	EG&G	78			FT	R	CDC CYPER 76	2.3	—
FAULTRAN	GE	77			FT	O	HONEYWELL 600/6000	60	—

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FAUNET	DEN-MARK	76			FT	O	PDP 8	—	—
FLASH 6/ FLASH 4	BAPL	74	1PL			R,RE	IBM 360/370 CDC 6600/7600	0.001 TO 100 RS	86 98
FOAM	B&W	73	2C			P	CDC 6600	—	—
FORE1 FORE2	GE	65		1PL		R	PHILCO-2000 GE-635 IBM 360	V	64 79 77
FRANCESCA BWR	JNRC ITALY	72		2C,F		O	IBM 360	0.006 TO 0.012 TS	—
FRANTIC-NRC	NRC	77			FT	R	IBM 370	V	61
FRAP-T	ANC	75	2F			R	IBM 360/370	60 RS	68
FTA	HEDL	75			FT	R	CDC 6600	20 TO 50	66
GAPCON-THERMAL-2	PNL	75		2F		R	CDC 6600	60 TO 120	74
GLUB 1	BAPL	69		2F		RE	CDC 6600	300	64
GO	EPRI	78			FT	P	CDC 6400	25	—
GRASS	ANL	71		2F		O	IBM 370	—	—
HAFMAT	GE	75			SS	R,RE	CDC 6600/7600	5	64
HAMBO	AEE	67			C,F	O	IBM KDF	— 150	—
HECTIC II	AGN	65			F,SS	O	IBM 7090	V	—
HETRAP	WG	77	2T			R	IBM 360	60	61
HOTPIN	HEDL	75		2F		O	CDC 6600	—	—
HUXY	EN	74	2C,F			P	CDC 6600	—	—

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IMPORTANCE	LLL	78			FT	R	IBM 370	1	61
INTEG/INSPEC	SRL	73			FT	R	IBM 360	2	57
LION4	KAPL	67		2T		RE	UNIVAC CDC 6600 IBM	V	-- 68 --
LOCTA	$\overline{W}$	72	2C			P	CDC 7600	—	—
LOFTRAN	$\overline{W}$	72		1PL		P	CDC 6600	—	—
LYNX1/ LYNX2	B&W	78			SS,T H,C	P	CDC 7600	30	—
MARCH	BCL	80	1PL			O,D	CDC 6600	400	—
MARVEL	$\overline{W}$	77		1PL		P	CDC 7600	—	—
MFAULT	PNL	77			FT	O	CDC CYBER 74	300	—
MICE	ANL	—		2CT		O,D	IBM 370	V	—
MOCARS	EG&G	79			FT	O	CDC CYBER 76	2 TO 3	—
MOCUS	ANC	75			FT	R	IBM 360/370	120	64
MOXY/MOD32	ANC	73	2F			R	CDC 7600 IBM 360	— 180	123
MO219 (FLOT1)	$\overline{W}$	68		2PL		R,RE	CDC 6600	V	64
MO555 (ACT-1)	BAPL	67	2CT			R,RE	CDC 6600	V	66
MUCHA 1/ MUCHA 2	BCL	71	2C,F			R	CDC 6600	200 TO 400 RS	72
NAIAD	AAEC	78	2PL			R	IBM 360	60	92
NATRANS 2	ANL	75		2PL		R	IBM 360/370	100 TO 200	64

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NOWA	WG	73	1PL			D	—	—	—
NURLOC-1.0	BCL	68	2C			R	CDC 6400	1200	66
ODYN	GE	78		1PL		P	CDC 7600	—	—
ORCENT	ORNL	76		2S		R	IBM 360/370	1	66
PAD	LANL	77		2C		R	CDC 6600/7600	—	—
PARCH	CE	74	2C,F			P	CDC 7600	—	—
PARET	INEL	72		2C,F		R	IBM 360	22	68
PARK1	BAPL	77		1PL		R,RE	CDC 6600/7600	0.2 TO 0.5 TS	112
PELEN	BAPL	73		2F		R,RE	CDC 6600 UNIVAC	2.5 TS	64 61
PEPSE	EI	79			T,H	P	CDC 6600	20 TO 35	—
PLENUM	SRL	72			SS, H, C	R	IBM 360	V	61
PREP/KITT	ANC	72			FT	R	UNIVAC IBM 360	-- 0.06	66 70
PROSA	ANL	78			FT	R,RE	IBM 370	60	64
PTH-1	KI	64	2CT			R	IBM 7094	360	70
PUMP	B&W	73		1PL		P	HYBRID	--	--
QUENCH	B&W	75	2C,F			P	CDC 7600	--	--
RADAR	B&W	73		2F		P	CDC 6600	0.6 TS	—
RAPP	KAPL	69	2PL			R,RE	CDC 6600	0.1 TO 5	57

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RAS	INEL	77			FT	O	CDC 7600	V	—
REFLOOD	B&W	74	1PL			P	CDC 7600	V	--
REFLUX	LANL	78	2F			R	IBM 360	62	64
RELAP4/ MOD5	ANC	69	1PL			R,D	CDC 7600 IBM 360	3600 28800	157 196
RELAX	EN	75	1PL			P	CDC 6600	—	—
RELBOT	CE	—		2PL		P	CDC 6600	—	—
RELO1	BAPL	71			FT	R,RE	CDC 6600	60	57
REPP	BNWL	70			SS, C, F	R	UNIVAC 1108	30	64
RETRAN	EPRI	78	1PL			P	CDC 6600 IBM 360	3600 28800	10000
REXCO-H	ANL	73	2C,T			R,RE	IBM 360	60	66
SAFTAC	UNI	79			FT	R	UNIVAC 1108	V	61
SATAN VI	W̄	74	1PL			P	CDC 7600	600 TO 6000	--
SATURATED BLOWDOWN I	KE	66	1PL			R	IBM 7099	5400	74
SCORCH-B2	JAERI	76	1C,F			O	--	--	--
SETS	SNL	78			FT	R	CDC 6600	30	66
SGNIII	CE	74		1S		P	CDC CYBER 70	--	--
SIRENE	FRANCE	75				O,D	--	--	--
SLAP	W̄	72	1PL			P	CDC 7600	—	—

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STEWPOT	UK AEE	77	1PL			O	—	—	—
STRIKIN-II	CE	74	2F			P	CDC 7600	—	—
TAFY	B&W	72			T,F	P	CDC 6600	—	—
THETA 1B	ANC	71	2F			R	IBM 360	1800	97
THI3D	ANL	76			SS,T, H,F,C	R	IBM 370	90	72
TILT	ITALY	73	2PL			O	UNIVAC 1108	—	—
TOPS	BAPL	68		2PL		R,RE	CDC 6600	20	61
TORC	CE	75			SS, C,F	P	CDC 6600	—	—
TRAC	LANL	77	1PL			O,D	CDC 7600	V	—
TRANS-FUGUE1	AI	67		2F		R,RE	IBM 7094	120 TO 240	61
UTSG	GRS	78		2S		O,P	IBM 360	—	—
WAMCUTS	EPRI	78			FT	P	CDC 7600	V	—
WHAM 6	INEL	75	2PL			R	CDC IBM	40 180	61
WRAP-EM	SRL	77	1PL			O,D	IBM	V	—
WREM TOODEE2/ MOD3	NRC	76	2F			R	IBM 360/370	360 TO 420	82
XTHETA	EN	75	2F			P	UNIVAC 1108	—	—
ZOCO V	WG	72	2CT			O	SIEMENS 4004	V	—
6 KILER	EI	74	2C,F			R	IBM 360/370	120	61

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4. CONCLUSIONS AND RECOMMENDATIONS

Computer codes have been developed to carry out the analysis and prediction of the effects of a postulated accident or failure phenomenon upon light water reactor (LWR) systems. Brief descriptions of the purpose, use, and availability of a large number of these codes have been assembled in this initial compilation. The primary intent of this compendium (or a derivative) is to provide, in a single source, descriptions of codes that can be used by the reliability or maintainability analyst to identify significant failure phenomena and the interdependencies among these phenomena that can affect LWR safety and safeguards systems performance. Once these failure phenomena are identified and their interdependencies become known, the reliability or maintainability analyst will then be in a position to formulate failure event sequences (and hence, failure scenarios) that are based upon previous analysis rather than upon intuition.

The impact of credible failure event sequences upon LWR systems can be evaluated using system performance codes that model and analyze the effects of these failure scenarios, in particular, as these effects have an impact upon the performance of safety or safeguards systems. For safety systems, analyses will reflect the accidental nature of the postulated failure phenomenon, whereas, for safeguards systems, analyses will reflect the deliberate nature of premeditated assaults upon protective components. Indeed, the primary difference between analyses of safety system performance and analyses of safeguards system performance is in the data that characterize the initial failure phenomena. In contrast to accident scenarios, scenarios arising from a deliberately caused failure can entail data, that is, component failure probabilities, with values as high as 1.0. As a consequence, much higher system failure probabilities can be generated in evaluations of safeguards system performance than in evaluations of safety system performance.

The effects codes described in this compendium treat the transients that develop as a result of a postulated failure in terms of variables (such as temperature and pressure) that characterize the changing state of the system (or subsystems). For the safety system analyst, these transients either are the cause of additional failure phenomena (conditionally linked phenomena) or are failure phenomena per se. Assuming that correspondences can be discerned between sets of critical values for state variables (such as temperature and pressure) on the one hand and delimitable failure phenomena on the other, then these codes can be used to identify and characterize the phenomenal linkages, or interdependencies, between a postulated failure event and subsequent (conditionally linked) failure events. By providing a description of each code in terms of

1. Systems or subsystems involved,
2. Postulated failure phenomenon, and
3. Subsequent failure phenomena,

much of the reliability or maintainability analyst's guesswork in specifying plausible accident event sequences could be eliminated.

A formalized technique for effects code selection might entail a matrix consisting of rows corresponding to failure phenomena (items 2 and 3 above) and columns corresponding to the subsystems involved (item 1 above). Intersections of rows and columns could be labeled with the applicable code or codes. For a specified subsystem, a number of codes would involve common failure phenomena. In particular, one of the subsequent failure phenomena of one code might be the postulated failure phenomenon of another code. These linkages would be made apparent by the matrix, and the details of the phenomenal interdependencies within resulting event sequences, if not already described, could be determined by execution of the appropriate codes. Of specific interest to the reliability and maintainability analyst in specifying plausible event sequences would be the generation of data characterizing the time intervals between conditionally linked events. These data, along with an assumption regarding the probability distribution for the time of occurrence of the initiating event (associated with the first postulated failure phenomenon), constitute input data for analyses of system reliability or maintainability.

Assuming that it is both possible and practical to develop a formalized technique for effects code selection, then the event sequences, and the details of the event sequences, that are made apparent by the selected codes and the execution of these codes can be treated through the use of appropriate event sequence modeling codes (system performance codes). Many system performance codes are available or under development (for example, SETS, PREP, KITT,* AND MAP**), but they are not equally effective or efficient in application to various combinations of event sequences and systems or subsystems under consideration. Therefore, a formalized means for selecting the system performance code that is appropriate to a particular combination (event sequence and subsystem) could be used to great advantage by the reliability or maintainability analyst. To this end, a second matrix could be constructed with rows corresponding to significant event sequences and columns corresponding to affected systems or subsystems. At the intersection of each row and column, the applicable system performance code could be identified. In this manner, a modeling approach that is both effective and efficient, i.e., that has sufficient latitude to handle the selected subsystem and accident event sequence without requiring excessive time for set-up and computation, could be selected.

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*See pages C-49 and C-67 of this compendium.

**For a description of MAP, see D. Engi, "Maintainability Analysis Procedure (MAP)," SAND80-2497, (Albuquerque: Sandia National Laboratories, to be published).

In its present state, this compendium falls short of providing the information required to construct the first matrix. This shortcoming can be attributed primarily to the inherent difficulty in prescribing the discrete failure phenomena that correspond to sets of critical values for state variables. Additional effort will be required to develop or delineate information that can be used in the construction of formalized approaches to effects (and system performance) code selection.

APPENDIX A

Nuclear Power Plant LOCA Code Descriptions

This section contains a code summary for each of the computer codes that have been developed and are being used to analyze the loss-of-coolant response of nuclear reactor primary systems and primary system components. Codes that analyze both large- and small-break LOCA transients are included. Also included in this section are the codes used to analyze the behavior of other transients as well as LOCA transients.

ADVENT

Purpose

ADVENT was developed to analyze the containment thermal effects of a LOCA in a Westinghouse ice condenser nuclear unit similar to the Duke Power Company's McGuire units.

Status

A. Development	Developed by the Duke Power Company, Charlotte, North Carolina.
B. Availability	Released to the National Energy Software Center (NESC) in 1972. Cost $67.00.
C. Utilization	Used to predict the ice condenser containment LOCA behavior for safety requirements at the McGuire nuclear generating units.

Problems Solved

1. Determines the steady-state radial temperature distribution corresponding to fixed inside and outside temperatures.
2. Determines the radial temperature distribution in the steel containment shell and the concrete reactor (shield) building during the post-LOCA transient.
3. Calculates the temperature and pressure in annulus between containment shell and reactor building during the post-LOCA transient.
4. Determines the capability of the annulus ventilation fan to quickly attain and maintain a vacuum in the annulus after a LOCA occurs.

Systems and Subsystems Simulated

Containment, reactor shield building, and ice condenser.

Modeling

The model is one-dimensional with heat transfer occurring only in the radial direction. All physical properties are independent of temperature except the density of the annulus air, which behaves as an ideal gas and exists uniformly at its mean mixed temperature. The annulus air exists between the concrete reactor building and the steel containment building. Circulation of refrigerated air in the ice-condenser air ducts ceases at the initiation of the LOCA. Leakage of air across the concrete reactor building or the steel containment shell is negligible. Thermal contact resistances are neglected.

Limitations

1. The system is divided into three regions: an upper region, a lower region, and a middle, or ice condenser, region.
2. Finite difference equations are solved to determine the temperatures at various nodes for both the steady-state and the transient condition.
3. No allotment is made for steam/air concentration variations (diffusion in time); constant temperature environmental properties are assumed, limiting application to small total temperature/pressure gradient transients.

Computer Operations

ADVENT is operational on the GE 635 Mark Z time-sharing system. Written in FORTRAN IV.

Reference

R. S. Krell and J. L. Kelly, "ADVENT Annulus Ventilation Program for Westinghouse Ice-Condenser Nuclear Unit," Duke Power Company, December 3, 1971.

Contacts

J. L. Kelly and R. S. Krell
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28201

[redacted]

ASCOT-1

Purpose

ASCOT-1 was developed to study the thermohydraulic behavior of a PWR core during the blowdown phase of a LOCA.

Status

A.	Development	Developed by the Japan Atomic Energy Research Institute, Tokai Research Establishment, Japan.
B.	Availability	Operational.
C.	Utilization	Forms part of a PWR emergency core cooling system evaluation model and has been compared with analytical results.

Problems Solved

Determines system pressure, density, velocity distribution (radial and axial), fuel rod temperatures, and sonic velocity.

Systems and Subsystems Simulated

PWR core with fuel rods, assemblies, and flow channels.

Modeling

1. The core is considered axisymmetric and is modeled with many axial and radial mesh points.
2. To determine the temperature response of the fuel, the core is divided into concentric annular subregions and a representative fuel rod is analyzed. This fuel rod is modeled by a radial mesh within the code.
3. Considers heat transfer by forced convection, nucleate boiling, forced convection vaporization, transition boiling, stable film boiling, pool film boiling, transition pool boiling, superheated vapor forced convection, and low pressure flow film boiling.
4. Considers subcooled liquid, two-phase fluid and single-phase steam, and the zirconium-steam reaction.
5. The method of characteristics for solving partial differential equations is extended and applied to solve the conservation laws in the axisymmetric, two-dimensional, homogeneous-equilibrium flow field.

6. Code inputs require core inlet and outlet fluid and energy flow along with geometrical data and boundary conditions.

Limitations

1. Has an adjustable dimension so that the number of mesh points is dependent upon computer capacity.
2. Requires input from a code that calculates PWR loop behavior.
3. Treats only the blowdown phase of a LOCA transient.

Computer Operations

ASCOT-1 is operable on the FACOM 230-75 computer. Written in FORTRAN IV.

Interface/Overlap

ALARM-P1 has been used to obtain input for ASCOT-1.

Reference

K. Kobayashi and K. Sato, ASCOT-1-A Computer Program for Analyzing the Thermo-Hydraulic Behavior in a PWR Core During a LOCA, JEARI-M-7917, September 1978.

BEACON/MOD2A

Purpose

The BEACON series of programs is designed to perform a best-estimate analysis of the flow of a mixture of air, water, and steam in a nuclear reactor containment system under LOCA conditions.

Status

A.	Development	Developed by EG&G Idaho, Inc., at the Idaho National Engineering Laboratory, Idaho Falls, Idaho.
B.	Availability	Released to the National Energy Software Center (NESC) in 1978. Cost $138.00.
C.	Utilization	Simulates two-component, two-phase fluid flow in complex geometries using a combination of two-dimensional, one-dimensional, and lumped-parameter representations for various parts of the system.

Problems Solved

BEACON/MOD2A contains mass and heat transfer models for wall film and for wall conduction and is suitable for the evaluation of short-term transients in PWR dry containment systems.

Systems and Subsystems Simulated

PWR nuclear reactor dry containment systems.

Modeling

The model has a two-component (air and water) and two-phase (vapor and liquid) capability, permitting simulation of complex nonequilibrium problems such as flashing, two-phase jets, and liquid deentrainment. Each phase is described by its own density, velocity, and temperature, as determined by separate sets of mass, momentum, and energy equations. The two phases are coupled by exchange parameters that model the exchange of mass, momentum, and energy between the two phases. Computations for single-phase flow are also obtainable without special treatment.

Limitations

1. Neither dimension for the variable nodal arrangement of a mesh should vary between adjacent cells by more than a factor of two.

2. Centers of adjacent cells must align in either the radial (abscissa) or vertical (ordinate) direction.
3. No mesh may have more than 50 interior cells in either direction.
4. Coupling of the mesh regions must be on a cell-to-cell basis.
5. A maximum of 20 mesh regions and 20 lumped-parameter regions are allowed per problem set-up.
6. Total number of cells (nodes) in all regions is limited to 2184.

Computer Operations

1. Designed for the CDC 7600. Written in FORTRAN IV.
2. NESC executed the sample problem in about 40 seconds on a CDC 7600.
3. Requires 110,000 (octal) words of small-core memory and 46,100 (octal) words of large-core memory for the largest of the seven sample problems and uses FORTRAN IV (98%) and COMPASS (2%) languages.

Interface/Overlap

The BEACON solution procedure is based on the solution scheme of K-FIX, a computer program developed at Los Alamos National Laboratory.

References

Roy A. Wells, BEACON/MOD2A A Computer Program for Sub-compartment Analysis of Nuclear Reactor Containment -- A User's Manual, CDAP-TR-051, March 1979.

National Energy Software Center (NESC) Note 79-48, BEACON/MOD2A Tape Description, NESC No. 767.7600, May 26, 1979.

W. C. Rivard and M. D. Torrey, K-FIX: A Computer Program for Transient Two-Dimensional, Two-Fluid Flow, LA-NUREG-6623, NRC-4, April 1977.

Contacts

C. R. Broadus, R. J. Doyle, S. W. James,
J. F. Lime, and M. Sahota
P.O. Box 1625
Idaho Falls, Idaho 83401

BLODWN-2

Purpose

BLODWN-2 describes the hydraulic transients within the reactor coolant system that are caused by a LOCA through an assumed rupture.

Status

A.	Development	Developed by Westinghouse Electric Corporation, PWR Systems Division, P.O. Box 355, Pittsburgh, Pennsylvania.
B.	Availability	Proprietary.
C.	Utilization	Used for the blowdown analysis of Westinghouse PWR power plants.

Problems Solved

Calculates the transient hydraulic response of the coolant within the system at the location specified by the user. Also, core material temperatures are calculated. The following output parameters are printed: time, pressure, mass velocity, fluid enthalpy, density, velocity of sound, fluid quality, temperature, core fuel and cladding temperature, and heat flow into the coolant from the core.

Systems and Subsystems Simulated

Primary system and core, including piping, pressurizer, pumps, etc.

Modeling

1. The hydraulic system is divided into subregions in which fluid flows mainly along the longitudinal axis. Each subregion may be regarded as an equivalent "pipe" in which the wetted parameter can be of any shape. The whole hydraulic system thus becomes a complex network of the equivalent pipes. Each subregion is described by mesh points.
2. The boundaries (junctions) of each pipe may consist of dead ends, interfaces between two or more pipes, a pump, or the pressurizer, or they may represent a system rupture location through which the blowdown takes place.
3. The solution of the conservation of mass, energy, and momentum is accomplished by the method of characteristics.

4. Assumes one-dimensionality of flow and homogeneity of the vapor-liquid interface. A degree of thermal nonequilibrium is specified through a special manner, in which the local velocity of sound within the vapor-liquid mixture is computed.

5. Calculates the subcooled transition and saturated portion of the blowdown.

6. Includes a restart capability.

Limitations

1. The system can be represented by a maximum of 120 pipes. Each pipe can contain 3 to 20 mesh points (nodes) except for the leakage element pipes which must have only 2 mesh points. The maximum number of mesh points is 2800.

2. Each equivalent pipe must contain the volume of the actual network. The total volume of the model must equal the true volume of the region.

3. Emphasis in the code is on the hydraulic phenomena of the whole system rather than a detailed description of the space-time thermal condition within the reactor core. Therefore, although the level of sophistication of the heat transfer routines is not very high, it is satisfactory for calculation of energy transfer to and from the fluid and its effect on fluid flow and pressure history.

4. Not applicable to small-break LOCA analysis in its present form due to the homogeneous equilibrium assumption.

Computer Operations

1. Operable on the CDC 7600, using the Scope operating system. Written in FORTRAN IV.

2. The machine memory storage requirements are proprietary.

3. Uses the (0,0) overlay for program loading.

4. For a problem comprising 94 pipe geometry segments and 998 mesh points, the central processor run time was 1,542 seconds for a 0.4-second calculation.

Reference

S. Fabic, Description of the BLOWDOWN-2 Computer Code, WCAP-7918, 2nd rev., October 1970.

[redacted]

BURST-1

Purpose

BURST-1 provides for the study of pressurized liquid immediately after a rupture occurs (LOCA).

Status

A.	Development	Developed by the Idaho Nuclear Corporation for the Westinghouse Astronuclear Laboratory, Pittsburgh, Pennsylvania.
B.	Availability	Released to the National Energy Software Center (NESC) in 1970. Cost $61.00.
C.	Utilization	Partial time derivatives for mass velocity, pressure, and enthalpy, obtained from the three conservation laws for mass, momentum, and energy, are approximated with difference equations.

Problems Solved

Pressure, mass velocity, and enthalpy are calculated at equally spaced mesh points along the length of the system. Forces on designated sections of the system can also be computed, with provisions for directional changes.

Modeling

Pressurized liquid in a cylindrical system and a one-dimensional representation, assuming uniform conditions throughout any given cross-sectional area.

Limitations

1. A maximum of 50 total cylinders and adapters are allowed.
2. A maximum of 2000 space-steps are allowed for the whole system.
3. A maximum of 5000 time-steps are allowed.
4. Only subcooled, frictionless, adiabatic cases can be considered.
5. Many mesh points have to be specified to accurately track the pressure wave propagation and the resultant forces on systems sections.

Computer Operations

1. Operable on the IBM 360/75. Written in FORTRAN IV.
2. For a large problem in which the number of space-steps times the number of time-steps equals ~25,000, the program run time is about 1 minute of CPU on the IBM 7044 computer.
3. Requires 32K of memory.

References

R. P. Rose, G. H. Hanson, and G. A. Jayne, Hydrodynamics Describing Acoustic Phenomena During Reactor Coolant System Blowdown, IDO-17254, July 1967.

G. A. Jayne, BURST-1 Program Description, P00410, December 1967.

Contacts

G. A. Jayne and G. H. Hanson
Idaho Nuclear Corporation
P.O. Box 1845
Idaho Falls, Idaho 83401

R. P. Rose
Westinghouse Astronuclear Laboratory
P.O. Box 10864
Pittsburgh, Pennsylvania 15236

CHEMLOC-2

Purpose

CHEMLOC-2 computes the extent of core heating and metal-steam reaction following a hypothetical LOCA, including emergency core coolant failure in a water-cooled reactor.

Status

A. Development	Developed by the Argonne National Laboratory, Argonne, Illinois.
B. Availability	Released to the National Energy Software Center (NESC) in 1969. Cost $61.00.
C. Utilization	Used to calculate the response of the fuel rods and core to LOCA conditions, loss of flow, and failure of emergency core coolant systems.

Problems Solved

Calculates the extent of core heating and metal-steam reaction. Treats the effects of heat production caused by decay heating in the fuel, steam-metal chemical reaction, and heat transfer from the core to the bottom and top grid-support plates and to the wall surrounding the core; also treats the effects of core movement and material transfer on reaching slumping temperature.

Systems and Subsystems Simulated

Reactor fuel rods and core.

Modeling

By means of mass and heat balances on an elemental section of a fuel rod, simultaneous iteration equations were developed for the temperature of the fuel, cladding, and gas, as well as for the thickness of cladding oxidized and the fraction of steam in the gas throughout the core. Iteration equations are also included for the temperatures of the bottom grid plate, top grid plate, and core wall. The core-slumping model is governed by input parameters, including the slump factor and temperature, the fall factor, the fuel-retained factor, and drip and steam-block age factors.

Limitations

1. A maximum of 25 annular axial zones, each consisting of an equal number of rods.

2. A maximum of 40 axial length increments.

3. Does not make any neutronic calculations.

4. Has many input parameters that require extensive *a priori* user knowledge of the problem.

Computer Operations

1. Operable on the CDC 3600; requires the minimum capacity of the CDC 3600 using the Scope operating system. Written in FORTRAN IV.

2. Run time for the sample problem is approximately 2 minutes, which includes the time required to load the program.

Interface/Overlap

NURLOC-1.0 (p. A-79), LOCTA (p. A-59), BOIL1 (p. B-17)

Reference

James C. Henson, John L. Anderson, and Richard O. Ivins, *Chemloc II, A Computer Program Describing the Core Heating and Clad-Steam Reaction for a Water-cooled Power Reactor Following a Loss-of-Coolant*, ANL-7361, April 1968.

Contacts

J. C. Henson, J. L. Anderson, and R. O. Ivins
Argonne National Laboratory
9700 South Cass Avenue
Argonne, Illinois 60439

[redacted]

CLAPTRAP

Purpose

CLAPTRAP was developed to analyze the pressure distribution in multicompartmented total containment when the primary circuit suffers a LOCA.

Status

A.	Development	Developed by the Steam Generating Heavy Water Reactor Development Division of the Atomic Energy Establishment, Wenfrith, Dorchester, Dorset, England.
B.	Availability	Operational.
C.	Utilization	Used for safety analysis and design of containment buildings.

Problems Solved

Determines the transient pressure and temperature distribution in the containment following a LOCA. Also calculates the temperature of the containment structure, including the walls.

Systems and Subsystems Simulated

Multicompartment containment structures and associated internals, including the spray flow.

Modeling

1. The containment is subdivided into compartments and the vent or flow paths which connect them.

2. The mass and energy flow into the containment can be calculated by the code or input by the user. The internal calculations resemble those for the STEWPOT code (p. A-123), while input data can be obtained from any blowdown code such as RELAP.

3. Allows for varying degrees of water entrainment in each compartment. Fluid deposited can either be considered in equilibrium with other components (steam, air) in the compartment or completely separated or can be assumed to fall to a puddle on the floor without any enthalpy exchange with the vapor phase. If the pressure drops below the boiling pressure, evaporation of the puddle to equilibrium is assumed.

[redacted]

4. Several releases can be described as occurring in each individual compartment or in several separate compartments.

Limitations

1. The number of compartments and vents connecting compartments is limited by computer size.
2. The existence of hydrogen and the concomitant possibility of explosive concentrations of hydrogen are not taken into account.

Computer Operations

Information not readily available.

Interface/Overlap

RELAP (p. A-95) or other blowdown codes can be used to supply input data. The methods and models of STEWPOT (p. A-123) have been incorporated into CLAPTRAP to calculate releases into the containment. CLAPTRAP-II is an update version of CLAPTRAP.

References

W. H. L. Porter, CLAPTRAP - A Code for Calculating Pressure Transients in total Containments Enclosing a Breached Water Reactor Circuit, AEEW-R965.

W. H. L. Porter, CLAPTRAPII A Code for Calculating Pressure Transients in the Interconnected Compartments of a Total Containment Enclosing a Breached Water Reactor Circuit, AEEW-R1108, March 1977.

COBRA

Purpose

COBRA is a general program developed for steady-state and transient thermal-hydraulic subchannel analysis of rod bundle nuclear fuel elements.

Status

A. Development	Developed by, and currently under improvement at, Battelle Northwest Laboratory, Richland, Washington. Current development version is COBRA IV.
B. Availability	COBRA IV-I was released to the National Energy Software Center (NESC) in 1978. Cost $72.00 (CDC) and $66.00 (IBM).
C. Utilization	Utilized extensively in LWR and more recently in liquid metal fast breeder reactor (LMFBR) thermal-hydraulic analysis. Used to predict coolant conditions in the semiscale reactor vessel during a simulated LOCA transient.

Problems Solved

Computes steady-state and transient coolant flow, enthalpy and pressure drops in subchannels of the rod bundle along with other coolant thermal hydraulic parameters. Predicts detailed fuel and coolant temperatures, fuel melting, local boiling, fuel/clad differential expansion and heat flux distributions.

Systems and Subsystems Simulated

Nuclear fuel rod bundle and associated coolant channels.

Modeling

1. A lumped parameter treatment of a rod bundle in which the flow section is divided into subchannels and then the subchannels are divided axially into control volumes.
2. Equations of mass, energy, and momentum conservation are written as integral equations for the control volumes. These integral equations are space-averaged and reduced to a finite difference form.
3. This integral treatment necessitates providing semiempirical data at control volume boundaries (friction factors, mixing coefficients, etc.).

4. The entire set of control volume equations is generalized using matrix operators to allow an arbitrary layout of subchannels and thus to allow a truly three-dimensional representation of the bundle.

5. Includes boiling, nonboiling and crossflow mixing effects. The problem is treated as a boundary value problem using as boundary conditions inlet enthalpy, inlet flow, inlet crossflow, and uniform exit pressure.

6. At present, the program handles two-phase flow, using a pseudo-homogeneous slip flow model.

7. Used for a wide variety of rod bundle fuel element configurations. Handles distorted bundles and arbitrary axial, radial, and circumferential heat flux distributions. Can handle bundles with partial flow blockage.

Limitations

1. A maximum of four radial nodes in each fuel pin.

2. Two-phase flow modeling needs improvement. The separated slip flow modeling is valid, provided that the separated phase has uniform properties on mass flux in regions it occupies in the subchannels. Breaks down for gross departure from pseudo-homogeneous flow, such as annular flow in which liquid at the walls has a lower velocity than entrained droplets.

3. The general area of transient code verification is very weak. There is, at present, little or no data available even on single-phase transient flow distribution in rod bundles.

Computer Operations

1. Operable on the CDC 6600, the CDC 7600 using Scope, and the IBM 370 using OS/370. Written in FORTRAN IV.

2. The CDC 6600 version requires 202,000 words of core, peripheral devices for input and output, and four scratch units. The IBM 370 version requires 250,000 bytes of core and batch operation.

3. For the IBM 370/195, run time for a problem with 4 pins, 14 pin sections, 8 coolant channels, 20 axial nodes and 11 radial nodes in each fuel pin sector is about 4 minutes for a transient of 40 time-steps in 4 seconds of real time. For the CDC 7600, running time for a 7-pin bundle with 12 coolant channels and 36 axial nodes is about 5 seconds.

Interface/Overlap

COBRA I, II, and III are earlier versions of COBRA IV-I.

COBRA3-M is operational at the Argonne National Laboratory on the IBM 360.

COBRA IV-I is operational at the Battelle Northwest Laboratory on the CDC 6600 and 7600.

COBRA III-C is operational at the Idaho National Engineering Laboratory on the CDC 7600.

References

COBRA-I: D. S. Rowe, Crossflow Mixing Between Parallel Flow Channels During Boiling, Part I. COBRA--Computer Program for Coolant Boiling in Rod Arrays, BNWL-371, Pt. 1, Pacific Northwest Laboratory, Richland, Washington, March 1967.

COBRA-II: D. S. Rowe, COBRA-II: A Digital Computer Program for Thermal Hydraulic Subchannel Analysis of Rod Bundle Nuclear Fuel Elements, BNWL-1229. Pacific Northwest Laboratory, Richland, Washington, February 1970.

COBRA-III: D. S. Rowe, Interim Report. COBRA-III: A Digital Computer Program for Steady-State and Transient Thermal-Hydraulic Analyis of Rod Bundle Nuclear Fuel Elements, BNWL-B-82, Pacific Northwest Laboratory, Richland, Washington 1971.

COBRA-III C: D. S. Rowe, COBRA-III C: A Digital Computer Program for Steady-State and Transient Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements, BNWL-1695, Pacific Northwest Laboratory, Richland, Washington, March 1973.

COBRA-III M: W. W. Marr, COBRA-III M: An IBM 360 Computer Code for Analyzing the Thermal-Hydraulics of Pin Bundles, ANL-8131 Argonne National Laboratory, Argonne, Illinois, March 1975.

COBRA IV-I: C. L. Wheeler et al., COBRA-IV-I: An Interim Version of COBRA for Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements and Cores, BNWL-1962, Pacific Northwest Laboratory, Richland, Washington, March 1976.

COCO

Purpose

COCO analyzes the containment pressure transients following a LOCA. A detailed examination is made of the nonlinear physical phenomena that occur within the containment during the transient.

Status

A.	Development	Developed by the Westinghouse Electric Corporation, PWR Systems Division, P.O. Box 355, Pittsburgh, Pennsylvania.
B.	Availability	Proprietary.
C.	Utilization	Used for the evaluation of containment designs for a wide spectrum of containment pressure calculation applications, including containment integrity design analysis and emergency core cooling system (ECCS) containment back pressure analysis.

Problems Solved

Determines the containment transient fluid pressure and temperature response and the containment wall transient temperature response to a LOCA and simulates the nonlinear time-dependent physical phenomena that occur in the containment during the transient.

Systems and Subsystems Simulated

PWR dry containment structures and engineered safety equipment.

Modeling

1. The containment air-steam-water mixture is divided into two distinct systems: The first consists of the air-steam phase and the second is the water phase in the containment pump.
2. The time-dependent conservation equations of mass and energy, together with steam-table relationships, are solved for each distinct system (steam-air and water). Thermal equilibrium between the steam and air is assumed. However, this does not imply continual thermal equilibrium between the steam-air mixture and the water phase.
3. Includes models for internal and external sprays, containment venting and pump-back ventilation fan coolers, and a sump water

recirculation cooling system. Options have been provided for recirculation cooling analysis (residual heat removal (RHR) system).

4. Requires that the discharge mass and energy flow rates through the rupture be supplied by the user in tabular form. This information is usually obtained from another code. User inputs are required for internal and external spray and vent flows.

Limitations

1. The heat transfer to the containment walls is considered only in the direction perpendicular to the surface. Only multilayered flat walls are considered.
2. The discharge flow flashes into steam and water phases during the two-phase portion of the transient. The water falls into the sump, while the steam rises into the steam-air mixture. No droplet formulation is considered.

Computer Operations

Operable on the CDC 7600. Written in FORTRAN IV.

Interface/Overlap

REFLOOD (p. A-91) supplies input data to COCO.

References

F. M. Bordelon and E. T. Murphy, Containment Pressure Analysis Code (COCO), WCAP-8326, July 1974.

A. Kolflat and W. Chittenden, "A New Approach to the Design of Containment Shell for Atomic Power Plant," Proceedings of American Power Conference, Vol. XIX (1957), p. 651.

COMPARE

Purpose

COMPARE performs LOCA and other transient subcompartment analyses of the thermodynamic conditions in volumes connected by flowing functions with provisions for heat and mass additions.

Status

A.	Development	Developed at the Los Alamos National Laboratory, Los Alamos, New Mexico.
B.	Availability	Released to the National Energy Software Center (NESC) in 1977. Cost $61.00.
C.	Utilization	Used to perform transient subcompartment analysis of nuclear power plants and the evaluation of emergency core coolant systems (ECCSs).

Problems Solved

Solves the thermal and hydraulic parameters within volumes and across junctions for the geometrical model specified.

Systems and Subsystems Simulated

Nuclear power plant primary systems and components.

Modeling

1. Volume thermodynamics and junction flows are for homogeneous mixtures of steam, two-phase water, any three perfect gases, or combinations of the preceding.
2. Critical vent flow is based on the Moody model. Subcritical flow is based on compressible orifice flow of an ideal mixture with inlet and exit losses, resulting in a polytropic flow process. Two-phase water with air flow is calculated and inertial effects are calculated on the basis of an incompressible subelement relationship.
3. Volume thermodynamics and junction flows are determined independently in a quasi-static explicit manner. Mass and energy inflow or outflow accounting for each volume is done first for each timestep, assuming vent flows are constant. Thermodynamic equilibrium is assumed and state points determined. The resulting conditions are assumed constant and are used to calculate new vent flows between volumes for the next time interval.

4. Flow with inertia is calculated on the basis of a solution of the momentum equation with constant density and flow areas.

Limitations

1. A maximum of 100 volumes, 200 junctions, and 5 mass and energy addition tables (blowdown sets).
2. The water temperature is restricted to the range 278 to 620 K (40° to 656°F).
3. Numerical output may show system-dependent variations, particularly for the small values printed during the early time-steps.
4. The code only handles homogeneous 20 flow.
5. The code only evaluates Moody choking, not the spectrum that can occur.
6. The thermodynamic equilibrium volume assumption requires small time-steps to ensure calculation accuracy.

Computer Operations

1. The code is operable on the CDC 6600 and CDC 7600, using Scope 3.4 and 2.1, respectively, and the IBM 360 and IBM 370 using OS/360 and OS/370, respectively. Written in FORTRAN IV.
2. The code requires 3500 octal words for the CDC 6600 or 7600 and 125,000 bytes for the IBM 360 or 370.
3. A two-volume, one-junction problem requires a run time of 0.002 second per time-step on the CDC 7600 and 0.012 second per time-step on the IBM 370/165.

References

R. G. Gido, R. G. Lawton, C. I. Grimes, and J. A. Kudrick, COMPARE: A Computer Program for the Transient Calculation of a System of Volumes Connected by Flowing Vents, LA-NUREG-6488-MS, September 1976.

F. J. Moody, "Maximum Flow Rate of a Single Component, Two-phase Mixture," Journal of Heat Transfer, Trans. of ASME, 87 (February 1965), p. 134-42. [Moody model.]

COMPERCII

Purpose

COMPERCII was written to analyze the core refill and reflood portion of a PWR LOCA.

Status

A.	Development	Developed by Combustion Engineering, Inc., Nuclear Power Division, Windsor, Connecticut.
B.	Availability	Proprietary.
C.	Utilization	Used for safety analysis of PWR power plants to satisfy the NRC licensing requirements.

Problems Solved

The refill-reflood portion of the transient begins at the time safety injection water starts to enter the downcomers and fall into the lower plenum. COMPERCII determines fluid, thermal, and hydraulic conditions within the reactor vessel, along with the thermal response of the core, and calculates containment pressures and temperatures.

Systems and Subsystems Simulated

Simulates the reactor vessel, including downcomer, lower plenum, core and upper plenum, and safety injection systems. The primary system fluid conditions are simulated as a lumped parameter.

Modeling

1. The reactor vessel is modeled with control volumes and flow paths.

2. The following behavior is modeled: core reflood rate, reflood heat transfer, initial fuel and cladding temperature distributions, the steam-water loss in the injection section of the cold leg, safety injection flow rates, zircaloy-steam reaction, temperature dependence of material and fluid properties, flow resistance which accounts for the fluid conditions in the external piping system, energy balance in the injection section to determine fluid conditions entering the inlet annulus, loop seal blowout, and time-dependent containment back-pressure.

3. A provision for spillage of coolant through the break is also provided.

Limitations

1. One volume for the downcomer and lower plenum up to the bottom of the active core.
2. A core bypass volume that represents guide tubes and core periphery flow paths.
3. The core coolant model treats the core as a single channel with three axial regions, subcooled water, two-phase saturated water and steam, and saturated steam with entrained water droplets.
4. The fuel model has provisions for subdividing the core into 12 radial zones and 20 axial increments. At each radial and axial position, there is one fuel node and one cladding node.
5. One upper plenum volume that receives fluid from the core and core bypass and passes fluid to the containment.
6. The containment is considered as volume with a lumped parameter model.

Computer Operations

Operable on the CDC 7600. Written in FORTRAN IV.

Interface/Overlap

COMPERCII is an updated version of PERC and can receive input from STRICKIN-II (p. A-125) and CE-FLASH (p. A-45).

References

COMPERCII-A Program for Emergency Refill-Reflood of the Core, CENPD-134, Combustion Engineering Inc., rev., August 1974.

Description of Loss of Coolant Calculational Procedures, Combustion Engineering, Inc., Windsor, Connecticut.

CONTEMPT-LT/026

Purpose

CONTEMPT-LT/026 predicts long-term behavior of water-cooled nuclear reactor containment systems subjected to postulated LOCA conditions. It is used primarily for BWR containments but can be used to analyze PWR and experimental containment structures.

Status

A.	Development	Developed by Aerojet Nuclear Corporation at the Idaho National Engineering Laboratory, EG&G Idaho, Inc., and is currently under improvement at both EG&G Idaho, Inc. and Energy Incorporated, Idaho Falls, Idaho.
B.	Availability	Released to the National Software Information Center (NSIC) in 1970. Updated versions are available. Cost $119.00.
C.	Utilization	Used to model standard BWR containment (pressure-suppression, wet-well) systems, including the GE MARK I, MARK II, and MARK III systems. Also has been used to model experimental pressure-suppression systems.

Problems Solved

Calculates the time variation of compartment pressures, temperatures, mass and energy inventories, heat structure temperature distributions, and energy exchange with adjacent compartments. Capable of describing the effects of penetration and small-crack leakage on containment response. Models are provided for fan cooler and cooling spray engineered safety systems. One to four compartments can be modeled, and any compartment except the reactor system may have both a liquid pool region and a vapor atmosphere region above the pool. Each region is assumed to have a uniform temperature, but the two regions may be different. The user determines the compartments to be used, specifies input mass and energy additions, defines heat structure and leakage systems, and prescribes the time advancement and output options.

Systems and Subsystems Simulated

In particular, the BWR wet-well pressure suppression system but also PWR dry-well containment systems.

Modeling

1. The initial conditions of the containment atmosphere are calculated from input values, and the initial temperature distributions through the containment structures are determined from the steady-state solution of the heat conduction equations.
2. The input water and energy rates are evaluated at the midpoint of a time interval and added to the containment system.
3. Pressure suppression, spray system effects, and fan cooler effects are calcuated using conditions at the beginning of a time-step.
4. Leakage and heat losses or gains, extrapolated from the last time-step, are added to the containment system.
5. Containment volume pressure and temperatures are estimated by solving the mass, volume, and energy balance equations. Using these results as boundary conditions, the heat conduction equations describing structure behavior are advanced using an implicit technique. The resulting heat transfer rates are used to correct the previous estimates of the water and energy storage in the containment volume, and the containment conditions are obtained by solving, for the second time, the containment balance equations.
6. The pressure suppression routines use the conditions at the beginning of a time-step to calculate both the initial expulsion of water from the vents and the flow through the vents. From the calculated flow rates, mass and energy are removed from the dry well and added to the wet well.

Limitations

1. A maximum of 20 pipe elements in normal (unfailed) pressure suppression vent piping.
2. A maximum of 50 flow elements in one segment of the horizontal vent pressure suppression system.
3. A maximum of 10 horizontal vents or branches in a vent.
4. Does not allow calculations of hydrogen build-up or explosions in containment.

Computer Operations

1. Operable on the IBM 360 using OS/360 MUT; CDC versions are under development. Written in FORTRAN IV.
2. Requires 430,000 base bytes of memory (IBM 360) and a Calcomp or SC 4060 plotter for graphical output.

3. Typical IBM 360/75 run time is 0.021 second per time-step advancement for a 90 mesh point heat structure and about 0.3 to 2 seconds per time-step advancement for pressure suppression and wet-well vent calculations.

Interface/Overlap

CONTEMPT4/MOD 2 is the PWR analysis version (p. A-31).

CONTEMPT-LT/022 and CONTEMPT-LT/025 are earlier versions.

References

L. L. Wheat, R. J. Wagner, G. F. Niederauer, and C. F. Obenchain, CONTEMPT-LT - A Computer Program for Prediction Containment Pressure-Temperature Response to a Loss-of-Coolant Accident, ANCR-1219, June 1975.

K. D. Richert, "BUFIO, A Subroutine to Permit FORTRAN Access to IOOP," Phillips Petroleum Note, February 1966.

A. J. Smith, "Standard Package Description," Calcomp Plotter Subroutines (S0010.0), Appendix D, January 1971.

W. H. Rettig, R. C. Young, and N. H. Marshall, UPD - A Program to Update Source Decks, PO1751, and Addendum.

R. J. Wagner, "STH20, A Subroutine Package to Compute the Thermodynamic Properties of Water," ANCR Note, 1975.

NRTS Environmental Subroutine Manual, ANC Document, December 1972.

W. J. Mings, Correction to Uchida Model of CONTEMPT-LT/025 Program, SRD-43-76, December 1976.

W. J. Mings, Fan Cooler Containment Model in CONTEMPT-LT/025, SRD-25-76, November 1975.

W. J. Mings, Version 26 Modifications to the CONTEMPT-LT Program, SRD-83-76, April 1976.

Contacts

L. L. Wheat and W. J. Mings
Idaho National Engineering Laboratory
EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, Idaho 83401

G. F. Niederauer
Energy Incorporated
P.O. Box 736
Idaho Falls, Idaho 83401

CONTEMPT4/MOD2

Purpose

CONTEMPT4 calculates the long-term behavior of multicompartment PWR containment systems and experimental containment systems subjected to postulated LOCA conditions.

Status

A. Development	Developed and under improvement by EG&G Idaho, Inc., at Idaho National Engineering Laboratory, Idaho Falls, Idaho.
B. Availability	Released to the National Energy Software Center (NESC) in 1979.
C. Utilization	Used to model PWR dry containments, subatmospheric containments, dual containments, and experimental containment systems in a detailed multicompartmental fashion.

Problems Solved

Calculates the time variation of compartment pressures, temperatures, mass and energy inventories, heat structure temperature distributions, and intercompartment mass and energy exchange based on user-supplied values for compartment descriptions, time-step and edit controls, and selected problem features. Analytical models available to describe containment systems include models for containment fans and pumps, cooling sprays, fan coolers, heat-conducting structures, sump drain, and PWR ice condensers. Optional automatic time-step control allows time-step sizes within limits dictated by the user to be determined.

Systems and Subsystems Simulated

PWR or experimental dry containment structures.

Modeling

1. Using initial thermodynamic and geometric conditions supplied by the user, each compartment is initiated to input values and a steady-state solution of the heat conduction equation is performed for each containment structure.

2. The time advancement proceeds in the following manner using discrete time-steps:

    a. The code calculates mass and energy transfer rates due to the effects of user-specified mass and energy additions, sump drain flow, cooling sprays, fans, and fan coolers.

b. An implicit calculation is made to determine intercompartment junction flow rates.
c. A predictor-corrector scheme estimates heat structure effects.
d. Mass and energy transfer rates are then used to update compartment conditions from the start of the time-step by solving the mass, energy, and volume balance equations, including the condenser ice melting thermodynamic balances.

Limitations

1. A maximum of 999 lumped parameter compartments.
2. A maximum of 999 input tables.
3. A maximum of 99 intercompartmental flow junctions.
4. A maximum of 99 inputs for most other containment features.
5. Does not allow for hydrogen build-up and explosions.

Computer Operations

1. Operable on the CDC 7600 using Scope 2.1 (96% FORTRAN IV, 4% Compass). A separate edition, designated 7600B, has been prepared for use on systems other than the CDC 7600.
2. Requires 160,000 (octal) small-core memory (SCM) and 100,000 (octal) large-core memory (LCM) on the CDC 7600.
3. Calcomp or FR80 plotting devices are needed for graphical output.
4. Program run time is dependent upon problem construction and input. The sample problem required about 30 seconds.

Interface/Overlap

CONTEMPT-LT/026 (p. A-27) is the BWR analysis version.

HEAT1 is incorporated into the code for heat transfer calculations.

References

L. J. Metcalfe, W. J. Mings, J. E. Hartman, and A. C. Crail, CONTEMPT4/MOD2, A Multicompartment Containment System Analysis Program, TREE-NUREG-1202, February 1978.

L. J. Metcalfe, D. W. Hargroves, and R. A. Wells, "Containment Accident Analysis Using CONTEMPT4/MOD2 Compared with Experimental Data," Transactions of the American Nuclear Society, Vol. 28 (June 1978), pp. 412-414.

[redacted]

"CONTEMPT4/MOD2, Tape Description and Control Information," NESC No. 818.7600, National Energy Software Center Note 79-50, May 31, 1979.

"CONTEMPT4/MOD2, Description of Transmittal Tape for Other Than CDC 7600 Use," NESC No. 818.7600B, National Energy Software Center Note 79-51, May 31, 1979.

Contacts

L. J. Metcalfe
EDS Nuclear Inc.
220 Montgomery Street
San Francisco, California 94104

D. R. Meier
EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, Idaho 83401

G. F. Niederauer
Energy Incorporated
P.O. Box 736
Idaho Falls, Idaho 83401

CONTRANS

Purpose

CONTRANS calculates the transient response of the containment and annulus buildings to LOCAs and main steam line break accidents.

Status

A.	Development	Developed by Combustion Engineering, Inc., Nuclear Power Division, Windsor, Connecticut.
B.	Availability	Proprietary.
C.	Utilization	Used as a design tool for nuclear plant containments. Also used for safety analysis calculations.

Problems Solved

1. Determines the pressure and temperature within the containment and the temperature of the containment structure during a LOCA or a main steam line break accident.
2. Used for design purposes to determine, for a specified containment maximum pressure and temperature, the containment volume and configuration and, for a specific containment configuration, the maximum pressure and temperature. Also determines containment response to various LOCA transients.

Systems and Subsystems Simulated

A dry PWR containment building, an annulus building, and a model for the nuclear steam supply system (NSSS).

Modeling

1. The containment is divided into regions, and each region has several nodes.
2. The model consists of mass, energy, and volume balances on the containment and annulus buildings. Code input data for mass and energy release to the containment following an accident are usually obtained from separate primary system LOCA codes. The mathematical equations are in finite difference form and are numerically integrated in time.

3. The fluids (air, hydrogen, steam, and liquid water) in the containment building are assumed to be homogeneously mixed and in thermal equilibrium. A model for steam and liquid dimension at the break is also included.

4. Includes active heat sinks such as containment sprays, fan coolers, and shutdown cooling heat exchangers as well as passive heat sinks such as containment walls and interior structures.

5. A variety of wall heat transfer coefficients are available either as internal code functions or as code inputs.

6. For long-term NSSS energy release after the NSSS has reached pressure equilibrium with the containment, a model of the NSSS is included. This model treats decay heat, reactor coolant system and steam generator fluid stored energy, and primary and secondary metal-to-coolant heat transfer.

7. The annulus building model includes the containment walls as a heat source and the annulus walls as a heat sink. The change in annulus building volume resulting from the thermal and pressure expansion of the containment vessel walls is modeled along with the annulus ventilation system.

Limitations

1. CONTRANS uses a four-region model. The three regions in the containment are Region I (the vapor space), Region II (the containment sump), and Region III (the NSSS used for long-term analysis). Region IV is the annulus between the containment vessel and the shield building (annulus building) wall.

2. Assumes no heat transfer to or from the containment sump, to the concrete behind the steel containment liners, and to the environment.

3. Assumes containment isolation for pressure calculations.

Computer Operations

Operable on the CDC 7600. Written in FORTRAN IV.

Reference

Description of the CONTRANS Digital Computer Code for Containment Pressure and Temperature Transient Analysis, CENPD-140, 140A, Combustion Engineering, Inc., and Amendment 1, 2 and 3, April 1974.

CRAFT

Purpose

CRAFT determines the transient response of an LWR to a LOCA transient for both large- and small-break LOCAs.

Status

A. Development	Developed by Babcock and Wilcox (B&W), Nuclear Power Generation Division, Lynchburg, Virginia.
B. Availability	Proprietary.
C. Utilization	Used for licensing calculations of LOCA transients for B&W nuclear power plants in accordance with NRC regulations.

Problems Solved

Calculates primary, secondary, and core fluid thermal and hydraulic conditions and material thermal behavior during a LOCA. Core fuel behavior is also calculated.

Systems and Subsystems Simulated

Complete nuclear reactor system including primary, secondary, and emergency core coolant (ECC) systems and core systems.

Modeling

1. Modeled with control volumes and flow paths.
2. The conservation of mass and energy continuity and equation of state are solved for each control volume, and the momentum equation is solved for each flow path between volumes.
3. User option on implicit or explicit solution technique.
4. Models the reactor core, primary coolant pumps, steam generators, and interconnecting piping in any configuration and operating mode. Capable of modeling any thermal-hydraulic system containing similar components.
5. The homogeneous equilibrium assumption is used with appropriate correlation for two-phase fluid.

Limitations

1. A maximum of 50 control volumes is allowed.
2. A maximum of 80 flow paths of which 30 may be core flow paths.
3. Components with different thermal-hydraulic characteristics must be simulated as different nodes.

Computer Operations

1. Operable on the CDC 7600 using Scope 2.0 and can be modified to operate on the CDC 6600. Written in FORTRAN IV/FORTRAN Extended Version.
2. Requires 120K (octal) memory words and two tape drives for restart capability.
3. Typical program run times are 0.01 CPU seconds per control volume per time-step.

Interface/Overlap

CRAFT-1 is an earlier version of CRAFT-2. FLASH 6 (p. A-45), FLASH 4 (p. A-45), and RELAP3 (p. A-95) perform similar calculations.

Reference

R. A. Hendrick, J. J. Cudlin, and R. C. Foltz, CRAFT2 - FORTRAN Program for Digital Simulation of a Multinode Reactor Plant During Loss-Of-Coolant, BAW-10092, BAW-10092 1st and 2nd rev., July 1974.

DINO

Purpose

DINO calculates the temperature in a fuel rod subject to a LOCA.

Status

A.	Development	Developed by the Danish Atomic Energy Agency, Ros, Roskilde, Denmark.
B.	Availability	Operational.
C.	Utilization	Used to calculate the response of a fuel rod to various LOCAs. Tested on a BWR similiar to Oyster Creek.

Problems Solved

Determines the transient temperature distribution in a fuel rod and the transient heat fluxes at all nodal locations within the fuel rod.

Systems and Subsystems Simulated

Nuclear fuel rod including pellet, gap, and cladding.

Modeling

1. Considers only one fuel rod and its associated coolant channel. The rod and channel are subdivided into nodes by a grid. The grid is specified by input quantities, and the grid size is completely independent of the program.
2. The calculation is performed in two dimensions by a transient finite difference technique.
3. Considers heat transfer in three regimes: boiling, transition, and film boiling. Critical heat flux is also determined.
4. The fuel rod heat conduction and fluid thermal-hydraulic calculations are performed separately and related to each other by conservation of energy during each time increment.
5. Considers variable gas-gap conductances, temperature-dependent properties, and two-phase slip flow.
6. Requires that the core pressure, mass flow, and enthalpy at the core entrance, mass flow at the core exit, and fuel rod decay power be supplied from another program.

Limitations

1. Assumes no dimension change in the core occurs.
2. No fuel rod failure modes are accounted for.
3. Does not treat the metal-water reaction.

Computer Operations

Information not readily available.

Interface/Overlap

Can take data directly from the BUSCH code.

Reference

H. Abel-Larson and L. Larson, Heating in a Reactor Fuel Element Rod Under Transient Conditions, Part II, DINO-Temperature During a Loss-of-Coolant Accident, Danish Atomic Energy Agency, 80-3, September 1, 1972.

ECCSA4

Purpose

ECCSA4 predicts the thermal and hydraulic behavior of a single fuel rod and its associated flow channel during a LOCA and the subsequent emergency core cooling (ECC) injection.

Status

A. Development	Developed by the Battelle Memorial Institute, Columbus Laboratories, Columbus, Ohio.
B. Availability	Released to the National Energy Software Center (NESC) in 1971. Cost $66.00.
C. Utilization	Used to analyze the fuel rod and associated flow channel response to LOCA conditions. The fuel rod behavior is determined for quenching by ECC injection. The code can also be used to determine fuel rod response of experimental blowdown facilities.

Problems Solved

The fuel rod thermal behavior, both axially and radially, is determined and associated channel fluid behavior, such as velocity, quality, etc., is determined.

Systems and Subsystems Simulated

Single fuel rod and associated flow channel in a reactor core.

Modeling

An explicit, finite-difference technique is used to solve the conservation equations of the fluid in the core channel and the heat conduction equations in the fuel rod. Fluid properties are determined from property tables incorporated into the code. Accounts for flow stagnation and flow reversals.

Limitations

1. The fuel rod can be represented by a maximum of 11 radial nodes and 24 axial segments.
2. Each axial segment has an associated flow channel surrounding the fuel rod for a maximum of 24 segments.

Computer Operations

1. Operable on the CDC 6400 using Scope 3.3. Written in FORTRAN IV.
2. Requires 110,000 (octal) bytes of memory with one tape for restart and one tape for plotting data, if desired.
3. Typical run time is about 150 to 200 times real time using 1-foot axial segments in flow channel. Doubling the number of axial segments will increase run time by approximately a factor of four.

Interface/Overlap

MUCHA 1/MUCHA 2 (p. A-71).

References

R. A Cudnik, ECCSA and MUCHA, Computer Codes for the Analysis of Emergency Core Cooling Systems, BMI1916, September 1977.

6400 Programming Bulletin, Number 11, March 21, 1977.

Contact

R. A. Cudnik
Battelle Columbus Laboratories
505 King Avenue
Columbus, Ohio 43201

[redacted]

EFLOD

Purpose

EFLOD was developed for the simulation of the heat transfer and hydrodynamics of a nuclear power reactor during the reflood phase of a LOCA.

Status

A.	Development	Under development at Rensselaer Polytechnic Institute, Nuclear Engineering Department, Troy, New York.
B.	Availability	Developmental.
C.	Utilization	Used to study reflood thermal-hydraulics during a LOCA transient. Calculations using EFLOD have been compared to FLECHT-SET experimental data.

Problems Solved

Calculates system parameters during the reflood stage of a LOCA. Determines pressure, coolant and fuel rod temperature, quality, void fraction, mass flow, core liquid level, etc.

Systems and Subsystems Simulated

Models in detail only the reactor vessel during reflood. The primary loop (intact and broken) is also represented.

Modeling

1. The reactor system is modeled by seven volumes, which represent the upper plenum and hot legs, the cold legs and inlet plenum, the downcomer, the lower plenum, and three core volumes. One core volume moves with the quench front during the reflood process.

2. Uses a homogeneous, two-phase flow model to predict the fluid conditions in the regions of the system other than the upper core regions.

3. Uses an unequal-velocity, unequal-temperature (UVUT) two-phase flow model in the upper core regions where homogeneous conditions are not expected to exist.

4. Uses a radial conduction heat transfer model that calculates the thermal conditions in the fuel, gap, and cladding.

5. The differential equations for the conservation of mass energy and momentum are solved by a Runge-Kutta integration scheme, using the Fehlberg method.

Limitations

1. Simulates only the reflood stage of a LOCA transient and performs detailed calculations for only the reactor vessel.
2. Assumes that the fluid at the core inlet is in the liquid phase and the fluid at the outlet is steam.
3. The simulation of the loop considers only flow resistance.

Computer Operations

Information not readily available.

References

R. R. Gay, "EFLOD Code for Reflood Heat Transfer," Nuclear Technology, 44 (July 1974) pp. 246-57.

J. R. Waring, E. R. Rosal, and L. E. Hockrecter, PWR FLECHT-SET Bl Data Report, WCAP-8431, VC-78a, 1974.

FLASH 6/FLASH 4

Purpose

FLASH 6 and FLASH 4 determine the transient response of water-cooled reactors or hydraulic systems to severe variable-pressure operation resulting from large- and small-break LOCAs.

Status

A. Development	Developed and under improvement by Bettis Atomic Power Laboratory, Westinghouse Electric Corporation, West Mifflin, Pennsylvania. FLASH, FLASH 2, and FLASH 5 are earlier versions. CE-FLASH and WFLASH are specific user versions.
B. Availability	Released to the National Energy Software Center (NESC) in 1974. Restricted to use by U.S. citizens for either U.S. Government or commercial nonmilitary application at a U.S. installation. Cost $86.00 (IBM) or $98.00 (CDC).
C. Utilization	Used for accident analysis for preliminary safety analysis reports (PSARs) and final safety analysis reports (FSARs) of Westinghouse PWR plants. Analysis of Semiscale experiments and loss of fluid tests (LOFTs) have been performed with the FLASH code for NRC standard problems. Comparisons of calculations and data are available through the U.S. NRC Public Document Reading Room.

Problems Solved

Determines the hydrodynamic history of the fluid within the piping of a reactor plant under accident conditions. Thermal analysis is also performed including fuel rod thermal response. The program can be configured to represent all reactor systems and secondary systems with detailed nodalization of the core region in PWRs.

Systems and Subsystems Simulated

Complete reactor systems, secondary systems, individual hydraulic systems, core regions, and individual components.

Modeling

Uses an implicit numerical integration technique to solve the one-dimensional mass and energy conservation equation for each control volume and the momentum conservation equation for the flow paths. Thermodynamic equilibrium is assumed. Heat sources and sites are provided in the case models, steam generator models, and plant metal heat capacity models. Pressure in each control volume is obtained from the equation of state of the fluid. Acoustic effects of both subcooled and saturated blowdowns are included. Fluid properties are generated from stored data.

Limitations

1. FLASH 6 is limited to a maximum of 30 pressure determining nodes, 60 flow paths, 80 pressure determining core nodes, and 20 parallel core flow path channels.
2. FLASH 4 is limited to a maximum of 20 pressure determining nodes and 40 flow paths.
3. Plant and core geometry is fixed in time; hence, solution is not valid for significant dimensional changes including the core.
4. Homogeneous fluid flow assumptions are used.

Computer Operations

1. Operational on the IBM 360 and 370 using OS/360 and OS/370, respectively, and on the CDC 6600 and 7600 using Scope 3.3 and Scope 1.1, respectively. Written in FORTRAN IV.
2. Requires 2,500,000 bytes of memory for the IBM 360 and 144K (octal) central memory and 200,000 (octal) ECS or LCM memory for the CDC 6600 and 7600.
3. Run time varies between 0.001 and 100 times real time depending on problem type and code configuration.

Interface/Overlap

FLASH, FLASH 2, and FLASH 5 are earlier versions.
SATAN performs similar calculations (p. A-111).
WFLASH and CE-FLASH have been modified by Westinghouse Electric Corporation and Combustion Engineering, respectively.

References

J. J. Beyer, W. D. Peterson, D. A. Prelewicz, and G. W. Swartele, FLASH-6: A FORTRAN-IV Computer Program for Reactor Plant Loss-of-Coolant Accident Analysis, WAPD-TM-1249, July 1976.

J. H. Murphy, J. A. Redfield, and V. C. Davis, FLASH-5: A FORTRAN-IV Program for Transient Simulation of a Reactor Plant with a Detailed Core, WAPD-TM-999, May 1973.

[redacted]

T. A. Porshing, J. H. Murphy, J. A. Redfield, and V. C. Davis, FLASH 4, A Fully-implicit FORTRAN IV Program for the Digital Simulation of Transients in a Reactor Plant, WAPD-TM-840, March 1969.

J. A. Redfield, J. H. Murphy, and V. C. Davis, FLASH-2, A FORTRAN IV Program for the Digital Simulation of a Multinode Reactor Plant During Loss-of-Coolant, WAPD-TM-666, April 1967.

S. G. Margolis and J. A. Redfield, FLASH, A Program for Digital Simulation of the Loss-of-Coolant Accident, WAPD-TM-534, May 1966.

C. J. Pfeifer, CDC-6600 FORTRAN Programming - Bettis Environmental Report, WAPD-TM-668, January 1967.

W. R. Cadwell, ed., Reference Manual - Bettis Programming Environment, WAPD-TM-1181, September 1976.

G. W. Swartele, FLASH 6 Compliance with American Standard FORTRAN, BAPL Note, October 1976.

IBM360/370 FLASH R Reactor Transients Simulation Program Listing, MDC N2267-094, Volumes 1 and 2, July 1974.

V. J. Esposito, K. Kesavan, and B. A. Maul, WFLASH-A. A FORTRAN IV Computer Program for Simulation of Transients in a Multi-loop PWR, WCAP-8200, rev. 2, July 1974 (proprietary) and WCAP-8261, rev. 1, July 1974 (nonproprietary).

Contacts

7600 FLASH 6
J. J. Beyer, W. D. Peterson, D. A. Prelewicz, and
G. W. Swartele
Bettis Atomic Power Laboratory
Westinghouse Electric Corporation
Box 79
West Mifflin, Pennsylvania 15122

360 FLASH 4
C. K. Herkstroeter and K. Geib
McDonnell Douglas Automation Company
Box 516
St. Louis, Missouri 63166

Gilbert Associates, Inc.
P.O. Box 1498
525 Lancaster Avenue
Reading, Pennsylvania 19603

FOAM

Purpose

FOAM predicts the consequences of a rupture of a core flooding line and computes the level-swell history for the core during a small-break LOCA transient.

Status

A.	Development	Developed by the Nuclear Power Generation Division of Babcock and Wilcox (B&W), Lynchburg, Virginia.
B.	Availability	Proprietary.
C.	Utilization	Used as part of the B&W evaluation model to determine small-break LOCA consequences for safety analysis reports (SARs). The code has been compared to tests performed by General Electric Company, Hitachi Corporation, and Westinghouse Electric Corporation.

Problems Solved

Predicts the level-swell history in the core during a small-break LOCA and also determines axial void distribution and steam production rate.

Systems and Subsystems Simulated

The core of a PWR.

Modeling

1. The core is divided into axial nodes.
2. The Wilson correlation between bubble rise and void distribution is employed.
3. The core inlet and outlet flows are considered equal.
4. Requires power shape, power level, pressure, inlet subcooling, and quiescent water level as input.

Limitations

1. The core is divided into 36 axial nodes.
2. The solution considers steady-state with small flow rates.

3. No metal-water reaction or effects are considered.

4. Considers only constant pressure, thus condensation and flashing effects are not considered.

Computer Operations

Operational on the CDC 6600. Written in FORTRAN IV.

Interface/Overlay

The inputs to FOAM in the B&W evaluation model are from the ELRAFT (p. C-7) code.

References

B. M. Dunn, C. D. Morgan, and L. R. Cartier, Multinode Analysis of Core Flooding Line Break for B&W's 2568-MW_t Internals Vent Valve Plants, BAW-10064, April 1973.

Amendment No. 12, Wm. H. Zimmer Nuclear Power Station PSAR, Docket 50-358, Cincinnati Gas & Electric Company, Cincinnati, Ohio, April 1970.

H. Ogasswara et al., Cooling Mechanism of the Low Pressure Coolant Injection System of Boiling Water Reactors and Other Studies on the Loss-of-Coolant Accident Phenomena, Hitachi Research Laboratory, Hitachi Ltd., Japan (1973).

J. P. Cunningham, O. J. Mendler, and R. W. Steer, Core Uncovering Test at 100 and 400 psi, Westinghouse Electric Company, Report No. DT-T&H-517, March 1973.

[redacted]

FRAP-T

Purpose

FRAP-T predicts response parameters of a water-cooled, zircaloy-clad, oxide fuel rod during a LOCA, power-coolant mismatch (PCM), or flow blockage accident.

Status

A.	Development	Developed by the Aerojet Nuclear Corporation, Idaho Falls, Idaho.
B.	Availability	Released to the National Energy Software Center (NESC) in 1975. Cost $88.00.
C.	Utilization	Considers the coupled effects of response parameters by iterating to converge at each time-step in a transient. Transient heat conduction is analyzed.

Problems Solved

Predicts the fuel and cladding thermal and mechanical response and the internal gas pressure responses.

Systems and Subsystems Analyzed

Single fuel rod in a light water-cooled reactor.

Modeling

The fuel rod to be analyzed is divided into axial and radial nodes. A one-dimensional, transient, radial heat conduction solution is used to determine the time-dependent radial temperature distribution at each axial node.

Limitations

1. A maximum of 20 axial nodes is allowed.
2. A maximum of 20 radial nodes is allowed.
3. Performs only a single-rod analysis.
4. Considers only thermal reactors and a zircaloy-clad fuel rod with water as coolant.

5. Accompanying fluid coolant conditions must be user-specified, eliminating the possibility of coolant characteristic changes with heat addition causing further fuel cladding interaction.
6. Very dependent upon the accuracy of the thermal-hydraulic boundary conditions specified.

Computer Operations

1. Operational on the IBM 360 and 370.
2. Run time varies with the time-step selected and number of nodes. For a LOCA, run time is approximately 1 minute IBM 360/75 CPU per second of real time.
3. Requires 450,000 bytes of memory for compilation, 350,000 bytes for execution.
4. Written in FORTRAN IV, except for a number of Idaho National Engineering Laboratory (INEL) environmental routines written in Assembly language.

Interface/Overlap

FRAP-T uses the INEL environmental subroutines.

FRAP-S is the steady-state fuel rod analysis code designed to supply realistic initial conditions to FRAP-T.

Reference

J. A. Dearien and L. J. Siefken, FRAP-T - A Computer Code for the Transient Analysis of Oxide Fuel Rods: Volume I - FRAP-T Analytical Models and Input Manual, Aerojet Nuclear Company Interim Report I-243-3-51.1, July 1974.

[redacted]

HETRAP

Purpose

HETRAP is a heat transfer analysis program developed for use in the evaluation of a LOCA. It can also be used to calculate the steady-state and transient temperature fields in fuel rods and electrically heated rods for given heat transfer coefficients.

Status

A. Development	Developed by S. Malang, Kernforschungzentrum, Karlsruhe, West Germany, under the sponsorship of the NRC Division of Reactor Safety Research.
B. Availability	Released to National Energy Software Center (NESC) in 1977. Cost $61.00
C. Utilization	Used primarily for the calculation of heat flux and heat transfer coefficients at the surface of an electrically heated rod during a fast transient, using measured temperatures in the rod.

Problems Solved

During the transient, the following time-dependent variables are measured: rod power or current, sheath temperature, temperature inside the heater element, and coolant pressure and temperature.

Systems and Subsystems Simulated

Nuclear or electrically heated fuel rod and associated flow channels.

Modeling

The measured values are used to calculate the time-dependent heat transfer coefficient, heat flux at the rod surface, surface temperature, and stored energy in the rod. The calculation is divided into two parts: the steady-state temperature profile and the transient calculation.

Limitations

1. A maximum of 10 cross sections is allowed.
2. A maximum of 50 nodes in each cross section is allowed.

3. A maximum of 50 materials in each cross section is allowed.

4. A maximum of 1,000 variable pairs in transient calculations is allowed as input.

5. Calculation accuracy is subject to the accuracy of the input parameters.

6. Best used to fit previously collected heat transfer experimental data; not used for mechanistic heat transfer calculations.

Computer Operations

1. Operational on the IBM 360/75 and IBM 360/91. Written in FORTRAN IV.

2. Requires approximately 260,000 bytes of storage, a tape unit, a card reader, and a printer.

3. Four sample problems executed by NESC took less than 1 minute, including compilation time, on an IBM 370/195.

Interface/Overlap

Assumes the availability of a CalComp plotting package and is related to THETA 1B (p. A-129).

Reference

S. Malang, HETRAP: A Heat Transfer Analysis Program, CRNL-TM-4555, September 1974.

Contact

Robert A. Hedrick
Engineering Technology Division
Oak Ridge National Laboratory
P.O. Box X
Oak Ridge, Tennessee 37830

HUXY

Purpose

HUXY performs a heat-up analysis of a fuel rod bundle in a non-jet pump BWR core during a large- or small-break LOCA. Two versions of the code exist: a best-estimate analysis and an evaluation model code with 10 CFR 50 Appendix K assumptions.

Status

A.	Development	Developed by the Exxon Nuclear Company, Inc., Richland, Washington.
B.	Availability	Proprietary.
C.	Utilization	Used as a design and evaluation tool for BWR reload fuel.

Problems Solved

Determines the transient fuel rod temperature distribution and temperature distribution in the fuel bundle. Various options exist for determining metal-water reaction, fuel rod ballooning, and cladding rupture.

Systems and Subsystems Simulated

Nuclear fuel rods and fuel bundles.

Modeling

1. The fuel rods and bundle are modeled and subdivided into axial segments.
2. The coolant condition and initial conditions are user supplied and are usually obtained from another code.
3. The code treats all rods and the canister at an axial plane of interest in a single fuel bundle. The code can be considered to be a three-dimensional time code in the sense that it predicts the radial temperature distribution in each rod in a two-dimensional array as a function of time.
4. Complete generality with regard to bundle geometry and symmetry, multiple rod types (hollow or solid), gap coefficients, and the surface heat transfer coefficient, all of which are supplied by the user.

5. The code has the following models:

   a. A generalized radiation model with rod-to-rod and rod-to-canister view factors. Changes in geometry are considered.

   b. The Exxon nuclear blowdown heat transfer model for non-jet pump BWR LOCA analysis.

   c. A one- and two-dimension rod rewet model in which the time to rewet is either internally calculated or user-supplied.

   d. A provision which allows the internal pellet power to be specified.

   e. A transient-time-dependent gap coefficient model with pellet-to-clad radiation. An optional-gap coefficient-versus-time table can be input by the user.

   f. An option for rod ballooning after a prespecified cladding temperature has been exceeded.

   g. An option for rod failure by calculation of inside cladding metal-water reaction after a prespecified cladding temperature is exceeded.

   h. An option for a probabilistic approach to rod ballooning and failure.

   i. An option for calculating oxygen penetration into cladding, including both sides of cladding, following rod failure.

Limitations

1. A maximum of 50 pins is allowed in a symmetric bundle section arranged in only a square array with a maximum of 18 rows or columns.

2. A maximum of 50 axial increments and input table points are allowed.

3. The code is specific to Exxon nuclear fuel bundles.

Computer Operations

Operational on the CDC 6600. Written in FORTRAN IV.

Interface/Overlay

HUXY is an extensive modification of the MOXY (p. 95) code to include specific detail for Exxon nuclear components.

References

L. S. Steves, S. F. Gaines, and J. D. Kahn, HUXY: A Generalized Multirod Heatup Code, XN-74-34, 2nd rev., October 1974.

L. S. Steves, T. L. Krupinski, S. F. Gaines, J. D. Kahn, and L. C. Worley, HUXY: A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option, User's Manual, XN-CC-33(A), July 1974.

[redacted]

LOCTA

Purpose

LOCTA calculates detailed thermal conditions in the core of a PWR power plant during a LOCA transient and determines fuel, cladding, and coolant response during the transient.

Status

A. Development	Developed by the Nuclear Energy Systems Division of the Westinghouse Electric Corporation, Pittsburgh, Pennsylvania. LOCTA-IV is the evaluation model code with 10 CFR 50 Appendix K assumptions, and LOCTA-R2 is the best-estimate analysis code.
B. Availability	Proprietary.
C. Utilization	Used for design and analysis of core internals. Has been used to perform licensing calculations for Westinghouse PWR power plants.

Problems Solved

The fuel, cladding, and average environment temperatures, degree of cladding reaction, fuel rod failure, and energy release for a particular radial core location at various axial locations as a function of time are computed.

Systems and Subsystems Simulated

Reactor core, fuel rod including pellet, gap and cladding, and associated coolant channel.

Modeling

1. The core is considered as a number of typical radial regions and subdivided into axial segments. In each region, an average fuel rod is considered for computational purposes and subdivided into radial nodes.
2. The program solves the finite difference conduction equation for the fuel rod and estimates the coolant condition from the energy balance equation.

3. The following models are provided:

   a. Decay heat generation can be user input or calculated by using the ANS infinite decay heat generation standard.

   b. Fuel cladding can be either stainless steel or zircaloy. When zircaloy is used, the zirconium-water reaction is considered using the Baker-Just correlation with the effects of steam-limiting accounted for.

   c. Cladding melt slumping and melt falling is considered with an input specified increase in the zirconium-water reaction.

   d. Single-phase connective cooling, nucleate boiling, transition boiling, and stable film boiling are accounted for in the heat transfer modeling. The effects of fuel rod to rod and steam radiation are considered. During reflood, two-phase heat transfer is considered with the effects of droplet entrainment modeled.

   e. The variation of rod internal gas pressure and cladding hoop stress can be calculated, accounting for pellet and cladding thermal expansion. When the calculated cladding stress is equal to the ultimate cladding strength, the rod is considered to burst.

   f. The codes require input describing the blowdown, refill, and reflood processes in the core, including information describing the time behavior of the pressure, flow, and inlet fluid temperature and quality. These data are obtained from other blowdown analysis codes.

Limitations

1. The code is divided into several typical radial regions. LOCTA-R2 can calculate the behavior of two or more regions at a time, while LOCTA-IV considers only one region at a time.

2. The fuel rod is divided into three radial fuel pellet nodes and one cladding node in LOCTA-R2, while in LOCTA-IV the maximum number of nodes is not specified.

3. The maximum number of axial segments is 20 in LOCTA-R2 and 30 in LOCTA-IV.

4. The LOCTA-IV code performs two-dimensional (axial, radial) heat conduction calculations, while the LOCTA-R2 code only considers radial conduction.

5. For both LOCTA-IV and LOCTA-R2, interaction between radial regions in the core is not considered.

Computer Operations

Operational on the CDC 7600. Written in FORTRAN IV.

Interface/Overlap

The SATAN (p. A-111) and FLASH (p. A-45) codes produce input data for the LOCTA codes.

References

W. A. Bezella, C. L. Caso, and A. C. Spencer, *LOCTA-R2: Loss-of-Coolant Transient Analysis*, WACP-7835, January 1972.

F. M. Bordelon et al., *LOCA-IV Program: Loss-of-Coolant Transient Analysis*, WCAP-8305, June 1974.

L. R. Baker and L. C. Just, *Studies of Metal-Water Reaction, III, Experimental and Theoretical Studies of the Zirconium-Water Reaction*, ANL-6548, May 1965.

MARCH

Purpose

MARCH calculates the thermal-hydraulic behavior of a water-cooled reactor during a meltdown accident involving LOCA and reactor transients.

Status

A.	Development	Under development by Battelle Memorial Institute, Columbus Laboratories, Columbus, Ohio.
B.	Availability	In development and verification stages; due for release.
C.	Utilization	Used in the analysis of the TMI-2 accident to determine core behavior and condition. Used by the NRC Probabilistic Analysis Staff.

Problems Solved

Calculates primary system heat transfer, mass and energy balances, system pressure, leakage rates, core failure and melt, reactor vessel failure, decomposition and penetration of containment vessel concrete floor, and containment thermal hydraulics.

Systems and Subsystems Simulated

Reactor primary system including core and containment system.

Modeling

1. The code is divided into several subroutines or subprograms, each of which calculates a specific phenomenon.
2. BOIL calculates core heat-up and meltdown and performs energy and mass balances for the primary system.
3. PRIMP calculates leakage of steam and water from the primary system and places an upper bound on the flushing rate.
4. EXITQ calculates metal structure and exit gas temperatures and determines enthalpy flows into containment.
5. PROPS calculates saturated steam and water properties.
6. ECC calculates pumped and core flood tank emergency core coolant (ECC) injection.

7. MACE calculates containment building thermal hydraulics.
8. HEAD calculates bottom head failure due to the combined effects of melting and overstressing.

Limitations

1. BOIL models the primary system as a single volume subdivided into water and steam spaces. The core is modeled with a maximum of 10 radial and 50 radial power zones.
2. EXITQ models a maximum of four structures, each of which is a one-node representation of, for example, a grid plate, upper plenum internals, or length of primary coolant pipe.
3. No model is present for the removal of steam from the primary system gas space by condensation on the ECC water injected.
4. MACE models the containment by 10 volumes connected in series with a maximum of 200 heat conduction nodes distributed between 15 structures.

Computer Operations

1. MARCH was developed for the CDC 6600 using Scope. Written in FORTRAN IV.
2. Program run time for a 15-hour transient similar to TMI-2 is about 400 seconds.

Interface/Overlap

BOIL (p. B-17) is incorporated into MARCH.

Reference

R. O. Wooten, R. S. Denning, and P. Cybulshis, Analysis of the Three Mile Island Accident and Alternative Sequences, NUREG/CR-1219, January 1980.

MOXY/MOD32

Purpose

MOXY/MOD32 performs a thermal analysis of a planar section of a BWR fuel element during a LOCA.

Status

A.	Development	Developed by Aerojet Nuclear Corporation and under improvement by EG&G Idaho, Inc. at the Idaho National Engineering Laboratory, Idaho Falls, Idaho.
B.	Availability	Released to the National Energy Software Center (NESC) in 1973. Cost $123.00 (CDC).
C.	Utilization	Part of the Water Reactor Evaluation Model (WREM) used by the NRC to perform audit calculations on BWR power plants.

Problems Solved

Determines the thermal response of the fuel rods, fuel bundle, and canister of BWR fuel and calculates the temperature distribution within the fuel bundles and canisters.

Systems and Subsystems Simulated

BWR fuel rods, fuel bundles, and canisters.

Modeling

1. MOXY/MOD32 employs models that describe heat transfer by conduction, convection, and thermal radiation and that describe heat generation by metal-water reaction and fission product decay. Models that consider fuel-rod swelling and rupture, energy transport across the fuel-cladding gap, and the thermal response of the canister are included. MOXY requires that time-dependent data for the power, normalized to the steady-state power, for the heat-transfer coefficient, and for the fluid temperature be provided as input during the blowdown process. Internal models provide these parameters during the heat-up and emergency cooling phases.

2. An implicit numerical solution of the one-dimensional Fourier heat conduction equation is used to compute heat conduction in fuel rods, solid inactive rods, and hollow inactive rods. Interchange

of thermal radiation among surfaces is calculated using the network method described by Sparrow and Cess (see References section).

Limitations

1. A maximum of 10 time-step sets and 10 print-interval sets are allowed.

2. In considering time-dependent heat sources, a maximum of 50 normalized power-time pairs are allowed.

3. In considering time-dependent convection, a maximum of 50 convection heat transfer time-dependent pairs are allowed.

4. In considering time-dependent fluid temperature, a maximum of 50 fluid temperature pairs are allowed.

5. Time-dependent fluid temperature, normalized power, and heat-transfer coefficient must be provided as input during the blowdown process.

6. The radiation heat transfer matrix is read from a direct access storage device.

7. The use of fairly large time-steps (greater than 1 second) is allowed.

Computer Operations

1. Operable on the IBM 360 using OS/360 and on the CDC 7600 using Scope. Written in FORTRAN IV and Assembly language.

2. Requires 300 bytes of storage (IBM) and 5 units in addition to standard input/output units.

3. For the sample problem, program run time is approximately 3 minutes of CPU on the IBM 360/195.

Interface/Overlap

MOXY/MOD 001 is an earlier version of MOXY/MOD2.

RELAP4/MOD5, WREM-TOODEE2/MOD3, and MOXY/MOD32 are part of WREM.

VIEWPIN is used in MOXY for radiation heat transfer calculation.

GAPCON-THERMAL 1 performs calculations similar to those performed by MOXY.

References

D. R. Evans, MOXY, A Digital Computer Code for Core Heat Transfer Analysis, IN-1392, August 1970.

D. R. Evans, The MOXY Core Heat Transfer Code: Model Description and User's Guide, PG-R-76-003, December 1976.

D. R. Evans, The MOXY Core Heat Transfer Program: View Factor Model Improvements, RE-E-77-114, February 8, 1977.

WREM: Water Reactor Evaluation Model, Rev. 1, NUREG-75/056.

E. M. Sparrow and R. D. Cess, Radiation Heat Transfer (Belmont, California: Brooks-Cole, 1966).

NRTS Environmental Subroutine Manual, ANC Document, December 1972.

MOXY/MOD032, ACC No. 551, 360B, Tape Contents Description, Argonne Code Center Programming Note 77-23.

Contact

D. R. Evans
EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, Idaho 83401

MO555/ACT-1

Purpose

MO555/ACT-1 predicts the pressure transient within containment that results from a major rupture in the primary system of a PWR as a function of time.

Status

A. Development	Developed by Westinghouse Electric Corporation at the Bettis Atomic Power Laboratory, West Mifflin, Pennsylvania.
B. Availability	Released to the National Energy Software Center (NESC) in 1967. Restricted to use by U.S. citizens for U.S. Government or commercial nonmilitary applications at a government installation. Cost $66.00.
C. Utilization	Has been used to predict the containment response of Westinghouse PWR plants to fulfill safety requirements.

Problems Solved

Predicts the pressure history of the containment volume during a large rupture in the primary coolant system of a PWR. The temperature distribution within various types of heat sinks is calculated as a function of time. Containment discharge rates may also be determined as a function of both time and internal pressure.

Systems and Subsystems Simulated

Containment vessel.

Modeling

The differential equations of energy, momentum, and mass conservation are solved by a forward differencing technique. Means are available for introducing water and heat into the containment vessel and specifying various heat sinks for energy absorption.

Limitations

1. The single containment volume may have a maximum of three regular inflow tables, one table for spray flow, 20 heat sinks, two

linear, and one log-log heat input table. The program also has two special inflow tables for the Westinghouse FLASH code output.

2. Only applicable to dry (free volume) containments.

Computer Operations

1. Operates on the CDC 6600 using Scope 2.0. Written in FORTRAN IV with batch output.
2. Program run time is dependent upon the amount and complexity of the input table and heat sinks.

Interface/Overlap

Accepts input to special tables directly from the FLASH (p. A-45) program output.

Reference

G. M. Flus, *ACT-1, a Digital Program for the Analysis of the Containment Transient during a Loss-of-Coolant Accident in a LWR*, WAPR-TM-693, May 1967.

Contact

G. M. Fuls
Westinghouse Electric Corporation
Bettis Atomic Power Company
West Mifflin, Pennsylvania 15122

MUCHA 1/MUCHA 2

Purpose

MUCHA 1 predicts the thermal and hydraulic behavior of any two reactor fuel rods and their associated parallel flow channel during large- and small-break LOCAs. MUCHA 2 provides the plenum fluid conditions as boundary conditions to MUCHA 1. MUCHA 2 predicts the thermal hydraulic behavior within the reactor vessel during a LOCA.

Status

A.	Development	Developed by Battelle Memorial Institute, Columbus Laboratories, Columbus, Ohio.
B.	Availability	Released to the National Energy Software Center (NESC) in 1971. Cost $72.00
C.	Utilization	Provides calculations of fuel rod performance and reactor vessel behavior during all phases of a postulated LOCA. Used for safety analysis of nuclear power reactors and experimental facilities.

Problems Solved

MUCHA 1 calculates the temperature response of two fuel rods and the fluid flow behavior in the associated flow channel. MUCHA 2 describes the thermal and hydraulic behavior of the primary coolant within the reactor vessel. A plenum model in MUCHA 2 has been developed to predict upper and lower plenum fluid conditions throughout the accident sequence including emergency core coolant (ECC) injection. The plenum conditions are used as boundary conditions in the MUCHA 1 program.

Systems and Subsystems Simulated

1. Fuel rod thermal and hydraulic behavior.
2. Vessel thermal and hydraulic behavior.
3. Upper and lower plenum fluid conditions.

Modeling

1. In MUCHA 1, an explicit, finite difference technique is used to solve the conservation equation in the parallel flow channel. The heat conduction equations for the fuel rods are solved by an explicit finite difference technique.

2. In MUCHA 2, an explicit, finite difference technique is used to solve the conservation equations for the parallel flow channel and the plena.

3. Thermodynamic properties are determined from property tables incorporated into the code.

Limitations

1. Allows up to a total of 15 radial nodes and 12 axial segments for MUCHA 1.

2. MUCHA 2 applies the above restrictions to each pair of equivalent fuel rods.

3. Channel cross-flow is not considered.

Computer Operations

1. Operational on the CDC 6600 using Scope 3.3. Written in FORTRAN IV.

2. MUCHA 1 requires 120K (octal) memory and one tape for restart capability.

3. MUCHA 2 requires 120K (octal) memory and two tapes, one for restart capabilities and one for storage of plena boundary conditions.

4. Program run time for MUCHA 1 is between 200 and 300 times real time using 2-foot axial segments. Doubling the number of axial segments will increase run time by approximately a factor of four.

5. Program run time for MUCHA 2 is between 200 and 400 times real time using 2-foot axial nodes. Doubling the number of axial nodes will increase the run time by approximately a factor of four.

Interface/Overlap

The MUCHA 1 program can accept plena boundary conditions from any system oriented code.

Reference

R. A. Cudnik, *ECCSA and MUCHA-Computer Codes for the Analysis of Emergency Core Cooling Systems*, BMI-1916, September 1971.

Contact

R. A. Cudnik
Battelle Memorial Institute
Columbus Laboratories
505 King Avenue
Columbus, Ohio 43201

NAIAD

Purpose

NAIAD computes the transient (including LOCA) and steady-state flow of two-phase compressible light water in a network of heated channels.

Status

A.	Development	Developed for the Australian Atomic Energy Agency (AAEC), New South Wales, Australia, and IBM 370/195 version sample problem by the National Energy Software Center (NESC), September 1977.
B.	Availability	Released to NESC in 1978. Cost $92.00
C.	Utilization	Mass, momentum, and energy are solved in each channel and the discharge flow is calculated intrinsically from the hydraulic equations.

Problems Solved

The one-dimensional conservation equations of mass, momentum, and energy are solved in each channel by a numerically stable, implicit finite-difference scheme. The network is then solved by an accelerated Gauss-Siedel iteration scheme.

Systems and Subsystems Simulated

Network of heated channels with two-phase compressible light water.

Modeling

A lumped fuel model as well as detailed heat transfer, thermodynamic equilibrium of the two phases, and the ANC pump model are included.

Limitations

1. Can be easily generated to handle any size problem, but the standard NAIAD code is limited to 50 axial nodes and 10 channels.
2. The Aerojet Nuclear Corporation (ANC) model is applicable only for small (25 hp), low-efficiency (35%) pumps. Larger or more efficient pumps cannot be represented accurately by the ANC pump model.

Computer Operations

1. Designed for the IBM 360. Written in FORTRAN IV and Assembly language.
2. Approximately 0.015 second per time-step per node are required on the Model 65 CPU; run time for the sample problem is 1 minute on the Model 65.
3. Requires 240K bytes of main storage and a 132-column printer.

Interface/Overlap

Four supplementary programs are provided: CONNOM blocks, ALIAS/NAME cards, SSGEN steam tables, and PUMPLOAD to load pump characteristic data.

References

G. D. Trimble and W. J. Turner, NAIAD - A Computer Program for Calculation of the Steady-State and Transient Behavior (including LOCA) of Compressible Two-Phase Coolant in Networks, AAEC/E378, April 1976.

G. D. Trimble and W. J. Turner, "Modification to NAIAD," AAEC Note, August 1977.

Contacts

G. D. Trimble and W. J. Turner
AAEC
Research Establishment
Private Mail Bag, Sutherland,
N.S.W. 2232
Australia

NOWA

Purpose

NOWA simulates the response of the primary system of a PWR during the low-pressure flow and refilling phase of emergency core cooling (ECC) after a LOCA.

Status

A.	Development	Under development by Laboratorium für Reaktorregelung und Anlegensicherung, Technische Universität, Munchen, West Germany.
B.	Availability	Developmental.
C.	Utilization	Used to study the ECC injection and refilling phase of a LOCA. Calculations have been performed which indicate parameters that affect the refilling rate of a typical PWR.

Problems Solved

Determines pressure distribution, flow rates, reflooding rate, metal temperatures, quality, steam production and condensation, etc.

Systems and Subsystems Simulated

Complete primary systems in PWR power plants including the piping, pumps, steam generation, the pressurizer, and the reactor vessel with associated internals.

Modeling

1. A multinode model (volumes and junctions) of the system geometry is employed.
2. Considers the following influences on the system behavior during the low-pressure flow and refilling phase:
   a. Blockage of flow paths and local flow resistance (injection points, pumps, etc.).
   b. Steam condensation at injection locations.
   c. Steam production in the core and evaporation of the ECC injected water at hot surfaces.
   d. Time-dependent behavior of water masses in the loops (water seals) and in the reactor vessel (downcomer, lower plenum, and core).

    e. Containment back pressure.
    f. Number and location of ECC injection points and time-dependent injection flow rate behavior.

3. Assumes homogeneous equilibrium two-phase flow. An option for nonequilibrium is under development.

4. Time-dependent conduction is considered to calculate fuel rod temperatures in connection with appropriate heat transfer coefficients.

Limitations

1. The multinode model can be arbitrarily chosen:

    a. The number of nodes and the connections between them,
    b. The number and location of injection points, as well as mass flows of injected fluid, and
    c. The location and size of primary system ruptures.

2. Uses a quasi-stationary solution (except in the steam generator and core) and, therefore, cannot handle very fast transient conditions.

3. Momentum transfer between steam and injected ECC water is neglected; however, condensation is considered.

Computer Operations

Information not readily available.

Reference

D. Brosche and R. Groberechter, "Simulation of Pressure Distributions and Steam Flow Processes During the ECC Low Pressure and Refilling Phase of a PWR," Topical Meeting on Water Reactor Safety, CONF-730304, Salt Lake City, Utah, 26 to 28 March 1973.

NURLOC-1.0

Purpose

NURLOC-1.0 performs core pressure vessel thermal analysis for a nuclear reactor LOCA.

Status

A.	Development	Developed at the Battelle Memorial Institute, Columbus Laboratories, Columbus, Ohio.
B.	Availability	Released to the National Energy Software Center (NESC) in 1968. Cost $66.00
C.	Utilization	Used to analyze reactor LOCAs and perform two-dimensional thermal analysis of either single pins or cylindrical shells.

Problems Solved

Performs core/pressure vessel thermal analysis.

Systems and Subsystems Simulated

Reactor vessel system.

Modeling

1. Performs a two-dimensional analysis of either single pins or cylindrical shells for a reactor vessel.
2. Uses an explicit finite-difference approximation.

Limitations

Allows two-dimensional analysis.

Computer Operations

1. Operable on the CDC 6400 using Scope. Written in FORTRAN IV.
2. Requires 65K or 32K memory with overlay.
3. A typical problem with 1,000 time steps requires about 20 minutes of run time on the CDC 6400.

Interface/Overlap

NURLOC-1.0 interfaces with ECCSA-I (p. A-41), a blowdown and emergency core cooling heat transfer program.

Reference

C. T. Walters and J. M. Gengo, *NURLOC-1.0, A Digital Computer Program for Thermal Analysis of a Nuclear Reactor Loss-of-Coolant Accident*, BMI-1807, July 1967.

PARCH

Purpose

PARCH is a pool boiling axial fuel rod and coolant heat-up code used in the analysis of small-break LQCAs.

Status

A.	Development	Developed by Combustion Engineering, Inc. (CE), Safety Analysis Group, Windsor, Connecticut.
B.	Availability	Proprietary.
C.	Utilization	Used for safety analysis of CE nuclear plants in conjunction with other analysis codes.

Problems Solved

Determines the transient fuel rod temperature during the almost quiescent core flow of a small-break LOCA and calculates core boil-off rates, clad rupture, and swelling.

Systems and Subsystems Simulated

Nuclear fuel rod and associated coolant subchannel.

Modeling

1. The fuel rod is subdivided into axial and radial segments.
2. The following inputs are required:
    a. Fuel rod and coolant subchannel dimensions,
    b. Axial fuel rod power shape in a form of normalized linear power at each axial node and power peaking factors,
    c. Transient two-phase coolant level in the core,
    d. Transient core pressure,
    e. Inner vessel liquid mass,
    f. The refill heat transfer coefficient, and
    g. Material property data (optional).
3. Fuel pellet and cladding dimensional changes are held constant with the exception of their effect on gap conductance.
4. The user may account for redistribution of decay heat from a hot rod to cooler surrounding rods via a gamma ray exchange between the rods.

Reference

PARCH - A FORTRAN IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup, CENPD-138, Combustion Engineering Inc., 1st rev., August 1974.

[redacted]

Reference

PARCH - A FORTRAN IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup, CENPD-138, Combustion Engineering Inc., 1st rev., August 1974.

[redacted]

PTH-1

Purpose

PTH-1 calculates containment shell pressure-temperature history resulting from aqueous coolant system blowdown.

Status

A.	Development	Developed by the Kaiser Industries Corporation, Oakland, California.
B.	Availability	Released to the National Energy Software Center (NESC) in 1964. Cost $70.00.
C.	Utilization	Used to calculate containment shell temperatures and thermal fluxes.

Problems Solved

Calculates temperature and thermal fluxes for a containment shell following a coolant system blowdown.

Modeling

1. The containment shell structure is represented as a group of slabs for which temperatures and thermal fluxes are found by solving difference equations that describe the slabs.
2. The contained gas pressure and temperature are calculated by using ideal gas law relations.
3. The slab values and the contained gas values are coupled in the solution by a forward difference method.

Limitations

1. Three materials are allowed for each slab.
2. The total number of slabs that can be represented is a function of the object program size and the number of nodes defining the temperature profile of each slab.
3. The results are dependent upon energy/time input supplied to the containment wall. This is not the same as discharge energy/time due to the time required for diffusion of the hotter vapor into the colder air at the containment wall.

Computer Operations

1. Designed for an IBM 7094 computer. Written in FORTRAN IV.
2. Requires 32K memory.
3. Run time for 11 slabs with approximately 50 nodes per slab requires about 6 minutes of IBM 7094 machine time to simulate a 24-hour period.

Reference

E. R. McLaughlin, J. S. Busch, and W. D. Harrington, PTH-1 Code Package IBM 7094, Pressure-Temperature History Program, June 1964.

Contacts

J. S. Busch, Kaiser Engineers
W. D. Harrington, Kaiser Engineers
Kaiser Industries Corporation
Kaiser Center
300 Lakeside Drive
Oakland, California 94604

QUENCH

Purpose

QUENCH was developed as a general program to simulate reactor core thermal transients for both large- and small-break LOCAs.

Status

A.	Development	Developed by the Babcock and Wilcox, Power Generation Group, Nuclear Power Generation Division, Lynchburg, Virginia.
B.	Availability	Proprietary.
C.	Utilization	Used to define variable sensitivities, for example, to determine the sensitivity of local metal-water reaction to local power for use in the whole core metal-water reaction determination.

Problems Solved

PARCH calculates:

1. Average fuel temperature in each region,
2. Average cladding temperature in each region,
3. Percent metal-water reaction in each region,
4. Heat rate to the fuel pin,
5. Integrated heat to the fuel pin,
6. Heat rate from the cladding, and
7. Integrated heat from the cladding as a function of time, unless otherwise specified.

Systems and Subsystems Simulated

Nuclear reactor core or any subdivision of the core and fuel rods within the core.

Modeling

1. The core is divided into regions of equal volume with provisions to allow any choice of power distribution. Each region contains a single fuel node and a cladding node.

2. The model includes heat generation from neutrons before reactor trip, delayed neutron heat, fission and activation product decay heat, exothermic zirconium-water reaction based on a parabolic rate law, heat transfer within the fuel rod, heat connection from the fuel rod cladding surface to any fluid within the core region, radiation heat transfer, and heat transfer from the fuel rods to steam necessary to sustain the metal-water reaction.

3. The effects of fuel rod melting and fuel rod slumping can be considered.

4. Heat from the zirconium-water reaction is completely generated in the cladding.

5. The program provides for steam rate limiting of the zirconium-metal reaction to any degree desired.

Limitations

1. A maximum of 50 core regions is allowed to simulate the entire core or any region of the core.

2. For the input tabular power option, the user must determine the portion of each heat source generated in the fuel on a case-by-case basis.

Computer Operations

Operable on the CDC 7600 using Scope. Written in FORTRAN IV.

Reference

B. M. Dunn and A. A. Mody, QUENCH-Digital Program for the Analysis of Core Thermal Transients During Loss-of-Coolant Accident, BAW-10106 and BAW-10106A, May 1975.

RAPP

Purpose

RAPP computes the relationship among mass, flow, pressure, and piping resistance (K-factor) for a system whose downstream pressure must be below the saturation pressure of the choked flow source fluid (LOCA-related).

Status

A. Development	Developed by Charles W. Sorenson for Knolls Atomic Power Laboratory, Schenectady, New York.
B. Availability	Released to the National Energy Software Center (NESC) in 1969. Restricted to use by U.S. citizens for either official U.S. Government or commercial nonmilitary application at a U.S. installation. Cost $57.00.
C. Utilization	Computes the parameters for the high velocity flow of a two-phase mixture of steam and water. The source fluid may be subcooled or saturated water, saturated steam, or a mixture of steam and water.

Problems Solved

RAPP uses the first and second laws of thermodynamics to either compute mass flow when pressures and resistances are specified for a constant-area flow path or compute pressure distribution in a piping run when flow is specified.

Systems and Subsystems Simulated

Specific applications include pressurizer surge line pressure drop and pressure distribution downstream of a relief valve.

Modeling

The major assumptions are

1. The differential loss of available energy is proportional to kinetic energy and differential increase in a geometric (K) factor,
2. The slip ratio is proportional to the specific volume ratio raised to a constant (input) power, and
3. Thermodynamic equilibrium is present.

Limitations

1. A maximum of six flow areas is allowed for computing pressure distribution with known flow.
2. 2000 pressure increments.
3. Applicable only to systems in which the area change to the minimum area point is gradual enough (preferably constant area) to allow thermodynamic equilibrium to be maintained.
4. The slip ratio between phases set as an input number x, i.e.,

   $(\text{Slip} = \frac{Ve}{Vg})^x$.

Computer Operations

1. Designed for use on a CDC 6600 computer. Written in FORTRAN IV.
2. Typical run time is 0.1 to 5 seconds per case.
3. The code machine requirements are 45K to compile and run.

Reference

Charles W. Sorenson, RAPP, A Digital Computer Program for Computing Flow Resistance and Pressure for High Velocity Flow of Steam-Water Mixtures, KAPL-M-6955, January 1969.

REFLOOD

Purpose

REFLOOD analyzes reactor coolant system behavior during the refill and reflood phases of the LOCA transient for both large- and small-break LOCAs.

Status

A.	Development	Developed by Babcock and Wilcox (B&W), Power Generation Group, Nuclear Power Generation Division, Lynchburg, Virginia.
B.	Availability	Proprietary.
C.	Utilization	Used for licensing calculation on B&W nuclear power plants in accordance with the NRC acceptance criteria for emergency core cooling (ECC) systems of light-water-cooled nuclear power reactors.

Problems Solved

Determines thermal and hydraulic parameters of the primary system coolant and thermal behavior of the primary and core material during the refill-reflood phases of a LOCA transient.

Systems and Subsystems Simulated

Nuclear reactor primary system, secondary system, and core region.

Modeling

1. The primary system including the ECC system is modeled with control volumes and flow paths and also represents the secondary side of each steam generator with a separate node. The core is represented by an average pin and average channel divided into axial segments.

2. The one-dimensional mass and energy conservation equations are solved for each control volume and the momentum equation is solved for the flow paths.

3. Handles two-phase flow with the homogeneous equilibrium model and accounts for two-phase friction and pressure drop.

4. Accounts for ECC water mixing and transport delay and water droplet entrainment within the core.

5. Accounts for heat transfer from the secondary to primary system during the transient.

Limitations

1. The primary system is modeled with 13 control volumes and associated flow paths.

2. The steam generator secondary side is modeled with one control volume and the primary side with two control volumes.

3. The core is divided into five equal-length axial segments for the purpose of heat transfer calculations.

4. Core energy is based on a table of normalized power versus time.

Computer Operations

1. Operable on the CDC 7600. Written in MIMIC digital simulation language.

2. Run time is dependent upon the problem and the length of the transient; however, typical run times are on the order of minutes.

Interface/Overlap

REFLOOD2 is an updated version of REFLOOD.

Reference

B. E. Bingham and K. C. Shieh, REFLOOD-Description of Model for Multinode Core Reflood Analysis, BAW-10093, July 1974.

REFLUX

Purpose

REFLUX calculates the temperature-time history of a representative fuel rod during the reflood stage of a hypothetical LOCA.

Status

A.	Development	Developed by Los Alamos National Laboratory, Los Alamos, New Mexico.
B.	Availability	Released to the National Energy Software Center (NESC) in 1978. Cost $64.00.
C.	Utilization	The logic used for selection of the appropriate flow regime for analysis of the cladding temperature transient is based on the axial position with regard to the continuous liquid level (based on a mass balance), a liquid carry-over criterion (derived from a force balance on a drop suspended in a vapor stream), and the local cladding surface temperature.

Problems Solved

A generalized boiling curve is constructed, and the local flow and clad conditions determine the applicable heat transfer coefficient.

Systems and Subsystems Simulated

A representative fuel rod during the reflood stage following a LOCA.

Modeling

1. The coolant thermal-hydraulic calculation in REFLUX is based on a single subchannel, one-dimensional, separated two-phase flow formulation solution of the conservation equations of mass, energy, and momentum.

2. REFLUX uses the fuel-clad temperature field from the preceding time-step to advance the coolant thermal-hydraulic solution to the present time-step. The void fraction in the flow boiling regimes is defined by the drift flux model.

3. Provision is made for analyzing nonequilibrium situations encountered under reflood conditions, and a complete two-step dispersed flow film boiling model is included.

Limitations

1. Reasonable results are for flooding rates of 0.4 to 10 in/s, pressures of 15 to 60 psia, peak powers of 0.5 to 1.5 kW/ft, inlet coolant temperatures of 50°F to saturation, and initial cladding surface temperatures of 250° to 2000°F.
2. Flow reversals cannot be analyzed.

Computer Operations

1. Designed for use on an IBM 360 computer. Written in FORTRAN IV.
2. NESC executed the sample problem in 62 seconds on an IBM 370/165.
3. The code machine requires a tape unit, a line printer, and 100,000 bytes of memory.
4. Should be compiled with the H-extended or the G1 Compiler to duplicate the author's results.

Reference

Walter Lee Kirchner, Reflood Heat Transfer in a Light Water Reactor, NUREG-0106, Volumes I and II, August 1976.

Contact

W. L. Kirchner, MS-603
Los Alamos National Laboratory
P.O. Box 1663
Los Alamos, New Mexico 87545

RELAP4/MOD5

Purpose

RELAP calculates the thermal-hydraulic behavior of a water-cooled nuclear reactor subjected to postulated transients such as large- and small-break LOCAs, pump failure, and power excursions.

Status

A.	Development	Developed by Aerojet Nuclear Corporation and under improvement by EG&G Idaho, Inc., at the Idaho National Engineering Laboratory, Idaho Falls, Idaho. Earlier versions are RELAP4/MOD4 and RELAP3. Versions under development are RELAP4/MOD6, MOD7, and RELAP5.
B.	Availability	Released to the National Energy Software Center (NESC) in 1969. Cost $195.00 (IBM) and $157.00 (CDC).
C.	Utilization	Used to calculate the accident response of a nuclear plant to fulfill requirements of preliminary satety analysis reports (PSARs) and final safety analysis reports (FSARs). Has been used to calculate the total system response of Semiscale and LOFT experiments along with various other experimental facilities. (Data and calculation comparisons are available through the NRC public document center.) RELAP4-EM is part of the NRC water reactor evaluation model (WREM).

Problems Solved

1. Calculates fluid conditions such as pressure, mass inventory, and temperature, and quality and thermal conditions such as temperature distributions, energy distributions, and heat fluxes in power generating and dissipating elements.
2. Provisions exist for calculations of reactor power, decay heat, and reactivity. RELAP4-REFLOOD and RELAP4-CONTAINMENT models are also available.

Systems and Subsystems Simulated

Total reactor plant system and subsystems, as well as individual components and systems, secondary systems, and core and fuel rods.

Modeling

1. The user supplies the geometric features of the system and the initial conditions, and the program solves the one-dimensional integral conservation and state equations by a fully implicit integration technique for each control volume and flow path. A homogeneous-equilibrium assumption is used.

2. RELAP4/MOD5 provides the following features: steam tables that contain derivatives, heat conduction between fluid volumes, momentum flux terms in the hydraulic equations, a pump model with inertia and friction, a limited air flow capability, a separate hot channel calculation, and core cross flow models.

Limitations

1. A maximum of 9 minor edit variables, 20 time-step cards, 20 trip control cards, 5 bubble parameter sets, 100 junctions (flow paths) between volumes, 75 control volumes, 12 pumps, 5 time-dependent volume descriptions on cards, 10 check valve types, 20 full system curves, 5 normalized leak-area-versus-time curves, 50 heat slabs, 20 heat slab geometrics, 7 heat slab materials, 50 core sections, and 20 data sets for heat exchanges without conduction are allowed.

2. Provides for fluid slip or two-phase flow transition.

3. Steady-state balances must be performed to ensure that the code is initialized properly prior to running a transient.

4. Provides many user-specified nodal options that require a thorough knowledge of the problem being analyzed to ensure accuracy.

Computer Operations

1. Operable on the IBM 360 using OS/360,370 and on the CDC 7600 using Scope. Written in FORTRAN IV (95%) and BAL (5%).

2. Requires 850,000 bytes of memory for the IBM 360 and 160,000 (octal) words for the CDC 7600, three tape drives, and a Calcomp plotter.

3. The graphics program PLOTR4M makes use of the SC-4060 integrated graphics system for microfilm output in addition to the Calcomp plotter.

4. Typical run times for a PWR LOCA analysis with 30 to 40 volumes is 15 to 60 minutes or more on the CDC 7600 and 2 to 8 hours or more on the IBM 360/75.

Future Efforts

Improvements to the break flow calculation, core heat transfer packages, and calculational techniques are under development. RELAP5 solves the two-dimensional conservation equation at certain locations, some two-component flows, and provides for improved core heat transfer.

Interface/Overlap

The following codes are either different versions of RELAP4/MOD5, improvements, or previous versions:

RELAP3/MOD62, MOD110,
RELAP4/MOD6, MOD7, and
RELAP5.

The codes SATURATED BLOWDOWN 1 (p. A-115) and NURLOC-1.0 (p. A-79) were initial blowdown codes from which RELAP evolved.

The codes FLASH 6/FLASH 4 (p. A-45) and SATAN (p. A-111) are improved versions of RELAP developed by the Westinghouse Electric Corporation.

References

RELAP4/MOD5, A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems, User's Manual, Volume I: RELAP4/MOD5 Description; Volume II: Program Implementation, Volume III: Checkout Applications, ANCR-NUREG-1335, September 1976.

RELAP4/MOD5 Update 2, ACC No. 369.7600B, Tape Contents Description, ACC Programming Note 77-3, October 29, 1976.

RELAP4/MOD5 Update 2, ACC No. 369.7600C, Tape Contents Description, ACC Programming Note 77-35, July 31, 1977.

RELAP4/MOD5 Update 2, ACC No. 369.360E, Tape Contents Description, ACC Programming Note 77-11, December 28, 1976.

W. H. Retting, "UPD - A Program to Update Source Decks," P01752; Addendum 1 to UPD by R. C. Young and N. H. Marshall; Addendum 2 to UPD by G. L. Singer; and Addendum 3 to UPD by W. H. Retting.

R. C. Young and S. D. Mathews, "PLOTR4M, RELAP4 Version 3 Plot Processing Program," ANCR Note, May 1974.

G. D. Grown, C. H. Bush, and R. A. Berman, The Integrated Graphics System for the SC4060: I. User's Manual, RM-5660-PR, December 1968.

Contacts

C. F. Obenchain, G. L. Singer (CDC 7600 contact), and J. E. Tolli (IBM 360 contact)
EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, Idaho 83401

RELAX

Purpose

RELAX determines the thermal-hydraulic response of a reactor system during the blowdown phase of LOCA transient. RELAX is a modification of the RELAP4/MODOOZ code.

Status

A.	Development	Developed by Exxon Nuclear Company, Inc., Richland, Washington.
B.	Availability	Proprietary.
C.	Utilization	Used to determine the blowdown response of the reactor system. The program has been applied to the blowdown analysis of the H. B. Robinson reactor system.

Problems Solved

Determines the thermal and hydraulic condition of fluid and core material during the blowdown phase of a LOCA transient.

Systems and Subsystems Simulated

A complete reactor system including core, piping, pumps, and steam generator, primary and secondary systems, or any portion of the system.

Modeling

RELAX is a modification of the RELAP4 code used to analyze the blowdown portion of a LOCA transient. (Consult the RELAP4 summary, p. A-95, for details.) The modifications are as follows:

1. Configured to give the evaluation model result in accordance with 10 CFR 50 Appendix K.

2. A signal has been added to indicate when the critical heat flux has been exceeded. After this point, the nucleate boiling correlation is not applied unless it predicts a lower heat flux than the transition boiling correlation.

3. The addition of a signal to indicate that the surface temperature has exceeded the local fluid saturation temperature by 300°F. After this point, the return to heat transfer coefficients predicted by the transition boiling correlation are precluded.

4. For qualities greater than 0.95, the pre-CHF heat transfer modes of nucleate boiling or forced convection vaporization are interpolated with the mode for steam cooling to give a smooth transition to superheated steam cooling.

5. For qualities greater than 0.95, a smoothing interpolation between the post-CHF heat transfer modes of film boiling is made, again to give a smooth transition to superheated steam cooling.

6. For stable film boiling, an alternate Groeneveld correlation, as given in the 10 CFR 50 Appendix K requirements, was substituted for the correlation used in RELAP4.

7. Heat generation due to the presence of the actinide elements uranium-239 and neptunium-239 was added to the heat generation as calculated in RELAP4. The actinide concentration is a function of the conversion ratio, which becomes an additional item in the program input list.

8. The fission product heating as determined in RELAP4 is essentially that of the ANS standard. In RELAX, it is multiplied by a constant factor of 1.2 to include the 20% uncertainty required by Appendix K.

Limitations

1. A maximum of 9 minor edit variables, 20 time-step cards, 20 trip control cards, 5 bubble parameter sets, 100 junctions (flow paths) between volumes, 75 control volumes, 12 pumps, 5 time-dependent volume descriptions on cards, 10 check valve types, 20 full system curves, 5 normalized leak-area-versus-time curves, 50 heat slabs, 20 heat slab geometrics, 7 heat slab materials, 50 core sections, and 20 data sets for heat exchanges without conduction are allowed.

2. Provides for fluid slip or two-phase flow transition.

3. Steady-state balances must be performed to ensure that the code is initialized properly prior to running a transient.

4. Provides many user-specified nodal options that require a thorough knowledge of the problem being analyzed to ensure accuracy.

Computer Operations

Operable on the CDC 6600. Written in FORTRAN IV. (The RELAP4 code to which modifications were made was operable on the IBM 370.)

Interface/Overlap

RELAX is a major modification of the RELAP4 code.

References

R. A. Pugh, RELAX-A Reactor System Blowdown Analysis Computer Code, XN-75-24, March 1975.

K. V. Morre and W. H. Rettig, RELAP4-A Computer Program for Transient Thermal Hydraulic Analysis, ANCR-1127, December 1973.

RETRAN

Purpose

RETRAN calculates and predicts nuclear power plant and associated systems behavior during postulated LOCAs, operation transients, and anticipated transients or other transient behavior for both PWRs and BWRs. The code can be adapted for use with many other fluid systems.

Status

A.	Development	Developed by Energy Incorporated for the Electric Power Research Institute, Palo Alto, California.
B.	Availability	Proprietary. Approximate non-utility user cost is $10,000 for tapes and manuals and $10,000 annually for updates. The code is free to utility users.
C.	Utilization	Has been used to predict the accident behavior of several nuclear power plants and to perform NRC standard problem calculations such as predicting the response of the LOFT and Semiscale facilities during both large- and small-break LOCA experiments.

Problems Solved

Calculates the thermal-hydraulic conditions within a nuclear power plant during transients. Calculates pressure, mass inventory, temperature, quality, temperature distribution, energy distribution, heat fluxes, etc.

Systems and Subsystems Simulated

Total reactor plant system and subsystems, individual components and systems, secondary systems, core, and fuel rods.

Modeling

1. Models the system geometry with volumes connected by junctions. The user supplies a complete description of the components and geometry of the system being analyzed.
2. Solves the one-dimensional fluid differential conservation equations and state equations for describing homogeneous equilibrium flow. Improvements for moving interface fronts are incorporated into the code.

3. The code includes

   a. An iterative steady-state initialization,
   b. A two-surface heat transfer model,
   c. Modified fills for time dependent P,h entry,
   d. Friction factors,
   e. Two-phase multipliers,
   f. Flow regime maps,
   g. A constant heat transfer coefficient option,
   h. An improved nonconduction heat exchanger model,
   i. Two-phase heat transfer correlations,
   j. Trip logic models,
   k. Control system models,
   l. A transport delay model,
   m. Nonequilibrium pressurizer models, and
   n. An auxiliary DNB model.

4. Uses a semimodular and dynamic dimensional structure so that future code developments resulting in new models may be tried and proven before they are incorporated.

5. Provides timing of feedback effects either from the core or a steam generator which is modeled with a transport delay model that considers movement of fluid through a region.

6. A variety of trip controls, control system models, and trip logic models have been included that can specify control functions typical of a reactor system and the sequence in which they are operated.

7. Uses an explicit time-step selection in which an iterative technique couples the time-step selection to the flow solution. Rapidly changing phenomena are sensed and the time-step adjusted. If flow is not changing rapidly, very large time-steps are used. This technique reduces the run time and guarantees a flow solution accurate to within a given tolerance supplied by the user.

Limitations

1. The model geometry (junctions and volumes) is limited by the size of the computer memory used in the calculations and the complexity of the input modeling.

2. Contains a one-dimensional homogeneous equilibrium model.

3. Does not contain specific models for reflood analysis and the secondary side turbine-generator.

Computer Operations

1. Operable on the Cyber 70/170 and CDC 6600 computers using Scope 3.4 or NOS/BE 1 and on the IBM 360 and 370 using OS/360 and OS/370, respectively. Written in FORTRAN IV.

2. Uses dynamic storage allocation; problem complexity dictates the amount of storage required.

3. The program run time is dependent upon the problem nature and model complexity. Typical program run time is about 15 to 60 minutes or more on the Cyber 76/170 and about 2 to 8 hours or more on the IBM 370/75 for a PWR LOCA analysis with 30 to 40 volumes.

Future Efforts

Development and implementation of a turbine-generator model.

Interface/Overlap

RELAP3/MOD110 and RELAP4 (p. A-95) are codes similar to RETRAN.

References

K. V. Moore, et al., RETRAN - A Program for One-Dimensional Transient Thermal-Hydraulic Analysis of Complex Fluid Systems, EPRI CCM-5, Volumes 1, 2, 3, and 4, December 1978.

N. H. Larsen, Core Design and Operating Data for Cycles 1 and 2 of Peach Bottom 2, EPRI NP-563, June 1978.

L. A. Carmichael and R. O. Niemi, Transient and Stability Tests at Peach Bottom Atomic Power Station Unit 2 at End of Cycle 2, EPRI NP-564, June 1978.

S. L. Forkner, D. L. Bell, and E. N. Winkler, Calculation of a Generator Load Rejection Transient in a 3293 NW BWR/4 with RETRAN-01-RET12B, Tennessee Valley Authority, July 1978.

E. N. Winkler, S. L. Forkner, and D. L. Bell, Calculation of a Feedwater Turbine Transient in a 3293 NW BWR/4 with RETRAN-01-RET12B, Tennessee Valley Authority, July 1978.

N. S. Burrell, G. C. Gose, J. F. Harrison, and G. R. Sawtelle, RETRAN Sensitivity Studies of Light Water Reactor Transients, EPRI NP-454, June 1977.

D. C. Poteralski, An Evaluation of the Uncontrolled RCCA Withdrawal Transient at Power Using the RETRAN Computer Code (Version 12D), Florida Power and Light Company, July 1978.

D. C. Poteralski, Comparison of the RETRAN Computer Code (Version 11D) Results With Measured Reactor Coolant Pump Coastdown Data, Florida Power and Light Company, July 1978.

Contact

L. J. Agec
Electric Power Reasearch Institute
P.O. Box 10412
Palo Alto, California 94303

REXCO-H

Purpose

REXCO-H (release 2) calculates the hydrodynamic response of primary reactor containments to a high-energy excursion.

Status

A.	Development	Developed and under improvement by the Reactor Analysis and Safety Division of the Argonne National Laboratory, Argonne, Illinois. REXCO-H (release 1) and REXCO-HEP are other versions of REXCO-H.
B.	Availability	REXCO-H (release 1 and 2) was released to the National Energy Software Center (NESC) in 1973. REXCO-H (release 2) is restricted to use by U.S. citizens for either official U.S. Government or commercial nonmilitary applications. Cost $66.00.
C.	Utilization	Has been used to predict the containment response of reactor plants. Comparison with experimental facilities has been made, and prediction of the Fast Flux Test Facility (FFTF) containment response has been performed.

Problems Solved

1. Calculates pressures and displacements of all nodal points at any instant of time and determines resulting damage to the reactor components.
2. Determines the propagation of shock waves and loads.

Systems and Subsystems Simulated

Containment structures and reactor vessel components.

Modeling

1. The two-dimensional hydrodynamic equations and equations of state of the reactor materials are expressed in Lagrangian form and set into finite difference equations.
2. Shock discontinuities are eliminated by the use of the Von Neumann-Richtmeyer pseudoviscosity factor Q.

3. Reactor vessels are treated as a thin-shell structure.

4. REXCO-HEP uses hydrodynamic, elastic, and plastic analysis.

Limitations

1. Allows for a maximum of 100 grid zones, 20 materials, 20 sections, 10 reactor vessels, and 50 points for the input stress-strain table of reactor materials.

2. Requires 1,000 cycles of the Calcomp display.

Computer Operations

1. Operable on the IBM 360/175 using OS/360. Written in FORTRAN IV.

2. Requires 700,000 bytes of core memory, two 9-track tape units, one 7-track tape unit, an IBM 2280 film recorder, and a Calcomp 780 plotter.

3. Run time for the sample problem was about 1 minute for 200 cycles of calculation.

Interface/Overlap

REXCO-H (release 1) is an earlier version of REXCO-H (version 2). REXCO-HEP is in the verification stage.

References

Y. W. Chang, J. Gvildys, and S. H. Fistedis, *Two-Dimensional Hydrodynamics Analysis for Primary Containment*, ANL-7498, November 1969.

"REXCO-H (release 2) Input Information Required," ANL-RAS Memorandum, October 26, 1971.

Y. W. Chang, J. Gvildys, and S. H. Fistedis, "Hydrodynamic Response of Primary Reactor Containment to High-Energy Excursion," *Nuclear Engineering and Design*, 12 (1970), pp. 344-360.

Y. W. Chang and J. Gvildys, "Dynamic Response of Reactor Containment to High-Energy Excursion," Proceedings of the 1st International Conference on Structural Mechanics in Reactor Technology, Berlin.

D. Basinger and J. Gvildys, *Three-Dimensional Representation of Surfaces by Computer*, ANL-7633, November 1971.

J. Von Neumann and R. D. Richtmeyer, "A Method of the Numerical Calculation of Hydrodynamic Shocks," *Journal of Applied Physics*, 21 (1950), pp. 232-237.

J. E. Ash and R. T. Julke, Comparison of a Two-Dimensional Hydrodynamics Code (REXCC) to Excursion Reactor Containment, ANL-7911, January 1972.

J. E. Ash, Y. W. Chang, and R. T. Julke, Comparison of a Two-Dimensional Hydrodynamic Code (REXCC) to Excursion Experiments for Fast Reactor Containment, ANL-9711, Supplement 1, July 1972.

Y. W. Chang, J. Gvildys, and S. H. Fistedis, "Analysis of the Primary Containment Response Using a Hydrodynamic-Elastic-Plastic Computer Code," Nuclear Engineering and Design, 27 (August 1973), pp. 155-175.

Contacts

Y. W. Chang, J. Gvildys, and S. H. Fistedis
Reactor Analysis and Safety Division
Argonne National Laboratory
9700 S. Cass Avenue
Argonne, Illinois 60439

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SATAN VI

Purpose

SATAN VI calculates the response of a nuclear steam supply system (NSSS) during the blowdown phase and part or all of the refill portion of postulated LOCAs.

Status

A. Development	Developed by the PWR Systems Division of the Westinghouse Electric Corporation, Pittsburgh, Pennsylvania.
B. Availability	Proprietary.
C. Utilization	Used for system design and safety evaluation calculations that are required for demonstrating safe recovery from a hypothetical LOCA in Westinghouse PWR plants. Developed specifically as part of the Westinghouse evaluation model that meets 10 CFR 50.46 Appendix K requirements.

Problems Solved

1. Determines the response of the total system to a blowdown and refill as a result of a LOCA.
2. Calculates fluid hydraulic parameters and the thermal behavior of core internal components.

Systems and Subsystems Simulated

Complete nuclear plant including pumps, pressurizers, steam generators, and core with internal components.

Modeling

1. The system geometry is represented by volume connected by junctions. Special volumes are considered for pumps, pressurizers, core, and steam generator tube regions. The fuel rods are subdivided axially in the core.
2. The code basically solves the space-time dependent momentum, energy, and mass conservation equations by a numerical integration technique.

[redacted]

3. The following models are incorporated into SATAN VI:

   a. All terms of the momentum conservation equation are considered,
   b. Two-phase flow effects are incorporated, including a two-phase pump model,
   c. Hot and average core channel flow including cross-flow effects are calculated,
   d. Clad swelling and rupture are included,
   e. Metal-water reaction effects using the Baher-Just correlation are considered,
   f. Fission product decay is assumed to be 1.2 times the ANS infinite operation decay standard. Decay of actinides are included for use in the reactor kinetics model,
   g. Emergency core coolant (ECC) effects are calculated, including accumulated bypass effects,
   h. The heat transfer model includes forced convection, subcooled and saturated (bulk) boiling, critical heat flux, transition boiling, and forced convection to steam, and
   i. Plant control parameters are available.

4. The input parameters describing the reactor coolant system provide the computer with the following information:

   a. A description of the way in which the volumes are interconnected,
   b. Specification of the function of each element, and
   c. Physical data for each element.

Limitations

1. Allows a maximum of 96 volumes with 3 special pseudo-volumes used for storage of plant control parameters.

2. Each volume can feed one or two other volumes and receive from as many as five volumes.

3. The core is modeled with up to 4 radial channels and 10 axial control volumes.

4. Equipped to analyze two-loop plants; three- and four-loop plants are analyzed by combining these loops into the two-loop model.

5. The problem terminates upon input specification or when the three following conditions are met:

   a. The downflow in the downcomer exceeds 100% of the ECC flow,
   b. The breakflow on the vessel side of the break is equal to zero, and
   c. ECC refloods the bottom of the core.

Computer Operations

1. Operable on the CDC 7600 using Scope 2.0. Written in FORTRAN IV.
2. Requires 135,000 (octal) words of small-core memory; the operating system requires 650,000 60-bit words of small-core memory and 256,000 60-bit words of large-core memory (CDC unit 7613). A card-reader photoelectric read, 1200 card/minute (CDC unit 407) and three units, 9-track tapes are required along with disk file storage and I/O hardware.
3. The code requires approximately the same run time as the RELAP4 code, which is about 10 to 100 minutes, depending upon problem specification and transient end time.
4. Has been used to calculate NRC Standard Problem No. 3.

Interface/Overlap

SATAN IV and V are previous versions of SATAN VI and do not incorporate the specific models and input required by 10 CFR 50.46 Appendix K.

The SATAN VI code provides mass and energy discharge rates (from the reactor coolant system to the containment) for the COCO and LOCTA codes, which perform containment pressure calculations.

References

F. M. Bordelon et al., SATAN VI Program - A Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant, WCAP-8306, June 1974.

F. M. Bordelon, "A Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant (SATAN-V Digital Code)", WCAP-7750.

Problem Specification for Standard Problem 3 - Gain-8-73, September 24, 1973, from Joseph M. Hendrie, W.S.A.E.C.

[redacted]

SATURATED BLOWDOWN 1

Purpose

SATURATED BLOWDOWN 1 calculates local thermal hydraulic parameters during the blowdown of the reactor primary coolant loop.

Status

A. Development	Developed by Kaiser Engineers, Oakland, California, and the Service Bureau Corporation, Palo Alto, California.
B. Availability	Released to the National Energy Software Center (NESC) in 1966. Cost $74.00.
C. Utilization	Used to predict thermal hydraulic parameters, temperature profiles, and heat fluxes in all wetted walls at each time increment. Has been used to predict the behavior of the Semiscale and LOFT experimental facilities during a simulated LOCA.

Problems Solved

Performs calculations of local pressures, mass flow rates, fluid qualities, specific volumes, temperatures, and local heat transfer coefficients at wetted walls at each time increment during a LOCA.

Systems and Subsystems Simulated

Primary coolant system.

Modeling

1. The primary coolant system is modeled by a number of spatial segments.
2. Three levels of interaction are performed for the calculations pertaining to each time interval:
    a. The first level of iteration involves the solution of the one-dimensional mass, energy, and momentum equations using the steam tables to formulate the equation of state of the fluid,
    b. The second level of iteration adjusts the flows at the branch tee, and
    c. The third level of iteration adjusts the flows at the rupture (exit) to meet the critical flow condition.

3. Temperature profiles and heat fluxes are calculated in all wetted walls at each time increment in order to obtain heat transfer coefficients.

4. The code is divided into a fairly large number of subroutines and functions so that model changes can easily be incorporated.

5. The input program model requires the specification of a heat exchanger, nuclear reactor core, a pump (associated with a check valve), a branch tee, and one "dead" segment, which is an off-branch of the loop.

Limitations

1. Allows a maximum of 53 spatial segments with each segment containing up to 4 different wetted surfaces.

2. All wetted surfaces are assumed to be slabs insulated on one side, except the reactor core fuel elements and heat exchanger tubing.

3. The pump has no effect during the transient.

4. Flow reversals are not permitted. The flow must be toward the branch tee, then to the exit.

5. The fluid is initially at or slightly below saturation temperature.

Computer Operations

1. Operable on the IBM 7099 using IBJOB, IBSYS version 12. Written in FORTRAN IV and MAP. MAP is used in steam table routines, error routines, and some input/output routines.

2. Run time is dependent upon the number of loop segments, the specified end pressure, the specified time-steps, and the exit area. Up to five time zones, each having a different time-step. can be specified. A 53-segment problem initially containing saturated water at 60°F and blowdown through a 16-1/2-inch inside-diameter exit (valve) requires about 90 minutes on the IBM 7099.

Interface/Overlap

SATURATED BLOWDOWN 1 is the predecessor to many BLOWDOWN analysis codes, including RELAP.

References

Stanislaw Fabic, Digital Computer Blowdown Analysis for Loss-of-Fluid Test Facility, Part 1: Engineering, 65-29-R, May 1965.

Digital Computer Blowdown Analysis for Loss-of-Fluid Test Facility, Part 2: Programming, Service Bureau Corporation, Palo Alto, California, July 1965.

[redacted]

SCORCH-B2

Purpose

SCORCH-B2 performs an analysis of core heat-up during a LOCA in a BWR plant.

Status

A. Development	Developed at the Japan Atomic Energy Research Institute, Tokai Research Establishment, Ibaraki, Japan.
B. Availability	Operational.
C. Utilization	Performs safety calculations to ensure that the BWR core keeps its coolable geometry throughout the postulated LOCA transient.

Problems Solved

Evaluates the peak clad temperature and the maximum oxidation thickness.

Systems and Subsystems Simulated

Core fuel assemblies, fuel rods, cladding, and channel box.

Modeling

1. The fuel assemblies are modeled by rows and columns of rods. The fuel rod is modeled with a radial mesh, and the axial zone being examined is divided into axial segments.
2. The fuel rods are classified into a number of small groups. All calculations are performed for representative fuel rods of each group.
3. Only the thermal behavior of the fuel is examined. Nuclear behavior of the fuel and the hydrodynamic behavior of the coolant, including heat transfer coefficients between the cladding surface and coolant, are given as input.
4. Radiant heat exchange between fuel rods and the channel box is considered. View factors are recomputed when deformation of the cladding occurs. The view factor is given as a function of the shadow area of radiation and irradiated bodies by parallel rays.

5. When the cladding temperature reaches a certain value, given as input, all rods in the group instantaneously balloon and rupture. After the deformation, the surface area, gap conductance, view factor, etc., are changed, and the oxidation reaction on the cladding inner surface is also considered.

Limitations

1. The maximum number of rows and columns in a fuel assembly is limited to three.
2. The maximum number of groups of fuel rods in the core is 64.
3. The maximum number of fuel pellet radial nodes is 10 with 1 node on the cladding inside surface and 1 node on the cladding outside surface.
4. The number of axial increments at the observation elevation is arbitrary.
5. Heat generation and transmission are only considered in the horizontal plane at a certain elevation of a fuel assembly.

Computer Operations

Information not readily available.

Reference

K. Abe and K. Sateo, SCORCH-B2 - Simulation Code of Reactor Core Heat-up During LOCA, JAERI-M6678, July 1976.

SLAP

Purpose

SLAP describes the behavior of a PWR nuclear power plant undergoing a small-break LOCA with a break size of up to 0.2 ft^2. The program applies to all closed-cycle LWRs.

Status

A.	Development	Developed by Westinghouse Electric Corporation, Nuclear Energy Systems Division, Pittsburgh, Pennsylvania.
B.	Availability	Proprietary.
C.	Utilization	Used for small-break LOCA safety analysis of PWR nuclear power plants. Analysis results have been compared with data from the Westinghouse Waltz Mill Test Facility.

Problems Solved

1. Calculates overall system behavior during a small-break LOCA.
2. Determines pressure, fluid temperature, system enthalpy, liquid and vapor mass, exit density and enthalpy, and fuel and metal temperatures.

Systems and Subsystems Simulated

Reactor primary coolant system, pressurizer, and steam generator.

Modeling

1. The system is subdivided in volumes or elements representing the primary system, pressurizer, and steam generator secondary.
2. The solution is based on the equations for conservation of momentum, energy, and mass as well as the equation of state. The model uses the homogeneous equilibrium assumption.
3. Break flow is calculated using the Fauske correlation for subcooled flow and the Moody correlation for saturated flow.
4. Residual and stored energy in the core is generated uniformly and transferred to the coolant proportional to the liquid and froth level in the core.

5. Provisions are provided for safety pump injection, accumulator tank injection, and secondary auxiliary feedwater. Steam generator and pressure variable vent and pressure relief valves are modeled along with the pressurizer heater.

6. A set of trips and controls can be modeled.

Limitations

1. The system is modeled by three volumes representing the bulk coolant and metal responses in the primary coolant system, pressurizer, and steam generator secondary.

2. The solution in actuality is a bulk energy balance on the three volumes and the break, assuming no flow and no pressure drop except between the primary system and the pressurizer.

3. The calculations are basically conservative in that core power and heat transfer and break flow are conservatively modeled.

Computer Operations

Operable on the CDC 7600. Written in FORTRAN IV.

References

G. L. Augustine, Small Loss-of-Coolant Accident Analysis for PWR Systems (SLAP Digital Computer Code), WCAP-7983, December, 1972.

F. J. Moody, "Maximum Flow Rate of a Single Component, Two-phase Mixture," Journal of Heat Transfer, Trans. ASME, 87 (February 1965), pp. 134-142.

R. E. Henry and H. K. Fauske, "The Two-phase Critical Flow of One Component Mixtures in Nozzles, Orifices and Short Tubes," Journal of Heat Transfer, Trans. ASME, 93 (May 1971), pp. 179-187.

STEWPOT

Purpose

STEWPOT performs an evaluation of the depressurization of the primary circuit of a water reactor and is particularly suitable for long-term release analysis.

Status

A.	Development	Developed by the Steam Generating Heavy Water Reactor (SGHWR) Development Division, Atomic Energy Establishment, Winfrith, Dorchester, Dorset, England.
B.	Availability	Operational.
C.	Utilization	Used for long-term estimates of the depressurization characteristics likely to be encountered during a LOCA with reactor depressurization.

Problems Solved

1. Determines system fluid parameters such as pressures, temperature quality, and break flow rates.
2. Can also determine fuel rod and cladding temperature distribution along with other metal temperatures.

Systems and Subsystems Simulated

Primary circuit of a reactor system; fuel rods with cladding can also be simulated.

Modeling

1. The system is modeled by a single volume and during a single mode. The fuel rod is modeled by a radial and axial mesh.
2. Emergency cooling water flow into the system is considered. Break flow is calculated from pressure and quality using the Moody correlation, with provisions for carryover.
3. Empirical heat transfer coefficients are used and come from experimental work for pressure transients with spray cooling.

4. Provisions for fuel-clad heat transfer are incorporated into the code, which allows for position-dependent heat rates and gas gap response.

5. Provision also exists for heat addition from other than fuel, metal surfaces.

Limitations

1. No limitations on the mesh size of the fuel and cladding model.

2. Does not consider flow dependencies and is essentially an energy balance on a static system accounting for inflows, outflow, and structure heat addition.

Computer Operations

Information not readily available.

References

W. H. L. Porter, *STEWPOT - A Code for Establishing Conditions in the Primary Circuit on Depressurizing a Water Reactor*, AEEW-M1099, March 1977.

F. J. Moody, "Maximum Flow Rates of a Single Component, Two-Phase Mixture," *Journal of Heat Transfer*, Trans. ASME, 87 (February 1965), pp. 134-142.

STRIKIN-II

Purpose

STRIKIN-II is used to calculate the core hot-spot transient clad temperature during the blowdown, refill, and reflood phases of both large- and small-break LOCAs.

Status

A.	Development	Developed by Combustion Engineering Inc. (CE), Safety Analysis Group, Windsor, Connecticut.
B.	Availability	Proprietary.
C.	Utilization	Used in the safety analysis of CE nuclear plants in conjunction with other safety analysis codes.

Problems Solved

1. Determines the fuel and clad transient temperature distribution along with the core hot spot.
2. Calculates fluid properties and heat transfer coefficients in the core.

Systems and Subsystems Simulated

Nuclear fuel rod and associated coolant subchannel.

Modeling

1. The fuel rod is subdivided into radial nodes, and the fuel rod and coolant subchannel are divided into axial increments.
2. The one-dimensional (axial) conservation of energy equation and the equation of state for the fluid are solved with provisions for local fluid expansion.
3. For the fuel rod, the one-dimensional (radially) cylindrical head conduction equation for each axial region along the rod is solved. The model explicity represents the gas gap region and dynamically calculates the gap conductance for each axial region.
4. Tabular time-dependent functions of flow rate, channel pressure, fluid enthalpy, normalized heat generation rate, and reflood heat transfer coefficients are required input for the code. The reflood heat transfer coefficient can be obtained from other codes.

[redacted]

5. Heat transfer is considered for forced convection to subcooled water and superheated steam using the Dittus-Boelter correlation, nucleate boiling using the Thom correlation, transition boiling using the correlation of McDonough, Milich, and King, stable film boiling using the Dougall-Rohsenow correlation, and pool film boiling using a CE proprietary formulation.

6. The zircaloy-steam reaction is considered using the Baker-Just equation.

7. Cladding rupture is also considered.

Limitations

1. The coolant channel and fuel rods can have a maximum of 20 axial nodes, 19 full-length nodes, and 2 half-length nodes at the subchannel beginning and end points. All fuel length nodes are of equal length.

2. The fuel rod may be divided into a maximum of 20 radial nodes.

3. The zircaloy-steam reaction has no provision for steam limiting.

Computer Operations

Developed for the CDC 7600. Written in FORTRAN IV.

Interface/Overlap

CE-FLASH (p. A-45) and COMPERC (p. A-25) are used in conjunction with STRIKIN to perform total system analysis.

References

STRIKIN-II - A Cylindrical Geometry Fuel Rod Heat Transfer Program, CENPD-135, NEV-01, Combustion Engineering, Inc., August 1974.

F. W. Dittus and L. M. K. Boelter, Heat Transfer in Automobile Radiators of the Turbular Type, University of California Publications, 2 (1930), pp. 443-461.

J. R. S. Thom et al., "Boiling in Subcooled Water During Flow Up Heated Tubes or Annuli," Proceedings Instrumentation of Mechanical Engineers, London, 180, Part 3C (1966), pp. 226-246.

J. B. McDonough, W. Milich, and E. C. King, Partial Film Boiling with Water at 2000 psia in a Round Tube, MSA Research Corporation, Technical Report 62 (1958).

[redacted]

R. L. Dougall and W. M. Rohsenow, *Film-Boiling on the Inside of Vertical Tubes with Upward Flow of the Fluid at Low Qualities* MIT-TR-9079-26 (1963).

[redacted]

THETA 1B

Purpose

THETA 1B computes the thermal response of a nuclear fuel rod subjected to a LOCA environment.

Status

A.	Development	Developed by Aerojet Nuclear Corporation at the Idaho National Engineering Laboratory, Idaho Falls, Idaho.
B.	Availability	Released to the National Energy Software Center (NESC) in 1971. Cost $97.00.
C.	Utilization	Used to determine thermal response of individual fuel rods or to model electrically heated test elements used in out-of-pile decompression experiments.

Problems Solved

Determines the temperature profile-history within a nuclear or electrically heated fuel rod and solves the energy equation for the associated fluid channel.

Systems and Subsystems Simulated

Nuclear or electrically heated fuel rods and associated flow channels.

Modeling

1. Solves the two-dimensional diffusion equation for a single fuel rod and the one-dimensional energy conservation equation for the associated fluid channel. The solutions are linked at the fuel rod-fluid interface by the boundary conditions of surface temperature and heat flux.
2. The energy equations in the fuel rod and fluid channel are solved by several numerical differencing techniques. Normally, the fuel rod equation is solved explicitly and the fluid equation is solved by the method of characteristics.
3. The heat transfer model considers forced convection in subcooled liquid and superheated vapor, nucleate boiling, forced convection vaporization, transition boiling, and stable film boiling. Various options are available for heat transfer correlations in different heat transfer regions.

Limitations

1. Solutions with small time-steps result in long computer run times.
2. Heat flux and coolant mass flow are user-specified boundary conditions.

Computer Operations

1. Operable on the IBM 360/75 using OS/360. Written in FORTRAN IV.
2. Requires a minimum of 400,000 bytes of core memory.
3. Run time is approximately 30 minutes for a 30-second transient.

Interface/Overlap

The numerical techniques employed in the CINDA3G program are used in THETA 1B.

References

C. J. Hocevar and T. W. Wineinger, Theta 1B, A Computer Code for Nuclear Reactor Core Analysis, IN-1445, February 1971.

J. D. Gaski, "Chrysler Improved Numerical Differencing Analyses for Third Generation Computers," Chrysler Corporation Technical Note, TN-AP-67-287, October 1967.

Contacts

C. J. Hocevar and T. W. Wineinger
Aerojet Nuclear Corporation
P.O. Box 1845
Idaho Falls, Idaho 83401

[redacted]

TILT

Purpose

TILT was developed to simulate the nuclear, thermal, and hydrodynamic behavior of a boiling water pressure tube reactor during transient conditions resulting from process perturbations and accidents (including LOCAs).

Status

A.	Development	Developed as part of the CIRENE project at the UNITA CIRENE-ENEL (Roma) and the CISE Laboratory, Segrate (Milano), Italy.
B.	Availability	Operational.
C.	Utilization	Used to simulate the performance of the CIRENE facility in response to operational or accident conditions.

Problems Solved

The transient quantities required for the characterization of the system behavior are described. These include reactivity, neutron flux, nuclear power, heat transfer coefficients, thermal power, temperatures, mass velocities, coolant enthalpies, pressures, coolant densities, and driving signals from control and safety loops as well as configurations of their actuators.

Systems and Subsystems Simulated

A boiling water pressure tube reactor and associated system and controls.

Modeling

1. The code is structured with a main program and eleven subroutines. Five subroutines perform the dynamic simulation of the various physical systems constituting the plant and the other six work out particular procedures.

2. Neutron kinetic is treated by a one-point monoenergetic approach which uses removal parameters and includes six delayed neutron groups from fission and three photoneutron groups.

3. Fuel thermal behavior is treated by subdividing the channels into axial elements. The temperature and radial power distributions are computed by applying an implicit finite-difference scheme in the space-time domain to the heat balance equation. The coolant behavior is determined by a lumped parameter model in which direct and reverse flows are considered. The possible states of the fluid are subcooled liquid, two-phase mixture, and superheated steam. In the two-phase mixture, the CISE-DIF correlation for pressure drop is used. The transient heat transfer model includes dry-out and rewetting.

4. Thermal hydraulic behavior of the coolant system is treated by a lumped parameter scheme. The homogeneous flow model is employed; the momentum balance equation is applied in a simplified form, neglecting the expansion terms. The subcooled liquid is considered an incompressible fluid. Thermal equilibrium is assumed.

Limitations

1. Considers two distinct power channels, each representative of the average conditions of a half core. An arbitrary number of axial elements can be selected.

2. The hydraulic model considers the following components: downcomers, the circulation pump, piping connecting the pump to the inlet headers, inlet headers, and emergency feed tanks.

Computer Operations

Operable on the Univac 1108 computer. Written in FORTRAN IV.

References

A. Magni, TILT: A Digital Simulator Programme for the Study of Hydrodynamic Processes and Core Heat of a Boiling Water Pressure Tube Reactor During Transient Conditions, Energia Nucleare, Vol. 20/B5/Maggio 1973.

M. Silvestri et al., The CIRENE Project, Paper delivered at the 4th International Conference on the Peaceful Use of Atomic Energy, Geneva, Switzerland, September 1971.

TRAC

Purpose

TRAC provides an advanced best-estimate predictive capability for the analysis of postulated accidents in LWRs. The postulated accidents currently being considered are large- and small-break LOCAs.

Status

A.	Development	Developed for the NRC by Los Alamos National Laboratory, Los Alamos, New Mexico.
B.	Availability	Operational in 1977. Soon to be released to the National Energy Software Center (NESC).
C.	Utilization	Determines the best-estimate response of LWRs and experimental facilities to accident conditions. Also used as a research tool for developing best-estimate methodology models and applications.

Problems Solved

1. Determines the hydraulic response of the system and its components and the thermal response of the system structure, including fuel and cladding.

2. Calculates pressures, temperatures, flow rates, qualities, density, reflood characteristics, fuel behavior, etc.

Systems and Subsystems Simulated

The entire reactor system including accumulators, pipes, pressurizers, pumps, steam generators, tees, valves, and vessels with associated internals.

Modeling

1. The physical system is modeled by mesh cells. The flow within ex-vessel components is treated in one dimension and a full three-dimensional (r,z) flow calculation is used within the reactor vessel. This is done to allow an accurate calculation of the complex multidimensional flow patterns inside the reactor vessel. For example, phenomena such as emergency core cooling (ECC) downcomer penetration during blowdown, multidimensional plenum and core flow effects, and upper plenum de-entrainment and fall-back during reflood can be treated.

2. A full two-fluid (six-equation) hydrodynamics approach is used to describe the steam-water flow within the reactor vessel, thereby allowing such important phenomena as countercurrent flow to be treated explicitly. The flow in the one-dimensional loop components is described using a five-equation drift-flux model, which differs from the standard four-equation drift-flux approach by the addition of a separate vapor energy equation. This provides a consistent nonequilibrium thermodynamic treatment in both the vessel and loop components and permits more accurate modeling of the fluid dynamics through a direct treatment of flashing and condensation effects.

3. The thermal-hydraulic equations in TRAC describe the transfer of mass, energy, and momentum between the steam-water phases and the interaction of these phases with the system structure. Since the nature of these interactions is dependent upon the flow topology, a sophisticated flow-regime-dependent constitutive equation package has been incorporated into the code.

4. TRAC incorporates a detailed heat transfer analysis capability for both the vessel and loop components. Included is a reflood tracking capability for both bottom flood and falling film quench fronts. The heat transfer from the fuel rods and other system structures is calculated using flow-regime-dependent heat transfer coefficients obtained from a generalized boiling curve based on local conditions.

5. TRAC has the ability to address entire accident sequences, including computation of initial conditions, with a consistent and continuous calculation. For example, the code models the blowdown, refill, and reflood phases of a LOCA.

6. The system of partial differential equations describing the two-phase flow and heat transfer is solved by finite differences. The heat transfer equations are treated as one-dimensional using a semi-implicit differencing technique. The fluid dynamics equations utilize either semi- or fully-implicit differencing under user control. The finite difference equations for hydrodynamic phenomena form a system of coupled, nonlinear equations that are solved by a Newton-Raphson iteration procedure.

7. Provisions exist for treating the following phenomena:

    a. ECC downcomer penetration and bypass, including the effects of countercurrent flow and hot walls,
    b. Lower plenum refill with entrainment and phase separation effects,
    c. Bottom flood and falling film reflood quench fronts,
    d. Multidimensional flow patterns in the core and plenum regions,
    e. Pool formation and fall-back at the upper core support plate (UCSP) region,
    f. De-entrainment and pool formation in the upper plenum,
    g. Steam binding,
    h. Average-rod and hot-rod cladding temperature histories,

i. Alternate ECC injection systems, including hot-leg and upper-head injection,
j. Direct injection of subcooled ECC water, without the requirement for artificial mixing zones,
k. Critical flow (choking),
l. Carryover during reflood,
m. Metal-water reaction,
n. Water hammer effects, and
o. Wall friction losses.

Limitations

1. All storage arrays in the code are dynamically allocated so the only limit on the size of a problem is the amount of core memory available. The number of reactor components in the problem, and the manner in which they are coupled together, is arbitrary.

2. Requires a long run time for large problems.

Computer Operations

1. Operable on the CDC 7600 using the standard CDC 7600 operating system with an FTN FORTRAN compiler and loader. Written in FORTRAN IV (compass matrix inversion routines are available but are not included).

2. Requires 600,000 words of small-core memory and 220,000 words of large-core memory.

3. Program run time is highly problem dependent and is a function of the number of total mesh cells and the maximum allowable time-step size. Total run time can be estimated from a unit run time of 2 to 3 ms per mesh cell per time-step and an average time-step size of 5 ms. Calculation of large problems (with several hundred mesh cells) requires the equivalent of at least several hours of CDC 7600 computation time.

Future Efforts

1. Work is underway to provide the capability to explicitly treat countercurrent flow between entrained droplets and falling films. The vessel hydrodynamics treatment has been augmented through the addition of a separate droplet mass equation and the required entrainment models. This modification will be made available in a future version of the code.

2. The basic flow equations are also being modified to include a noncondensible gas field. This will allow the code to account for the presence of accumulator nitrogen or air ingression from the containment during a LOCA.

3. The necessary components to model BWRs, e.g., jet pumps and steam separators, will be made available in a future version.

4. A fuel-cladding structural dynamics model will be added to provide more accurate gap conductance and cladding deformation effects.

5. Improved models for upper plenum entrainment/de-entrainment and UCSP fall-back will be developed as better experimental information becomes available.

6. The capability to treat other accident types such as Reactivity Initiated Accidents (RIAs) and anticipated transient without scram (ATWS) will be added.

Interface/Overlap

TRAC-P1A is a version of TRAC written for PWR large-break LOCAs.

Reference

TRAC-P1A, An Advanced Best Estimate Computer Program for PWR LOCA Analysis, NUREG/CR-0665, LA-7777-MS, (Los Alamos: Los Alamos Scientific Laboratory, May 1979).

WHAM 6

Purpose

WHAM 6 is used to calculate pressure, velocity, and force transients in liquid-filled piping networks during both large- and small-break LOCA transients.

Status

A. Development	Developed at the Idaho National Engineering Laboratory and under improvement by EG&G Idaho, Inc., Idaho Falls, Idaho.
B. Availability	Released to the National Energy Software Center (NESC) in 1975. Cost $61.00.
C. Utilization	Used to calculate water-filled system LOCAs or variable pressure response. Has been used to calculate the subcooled liquid response of the Semiscale and LOFT facilities during simulated large- and small-break LOCA experiments and showed good agreement when compared to fast response pressure measurements.

Problems Solved

1. Calculates pressure, temperature, velocity, and forces resulting from a water hammer caused by various transients.
2. A transient can be initiated either by closure or opening of one or more exit paths (equivalent to system rupture) or by a prescribed gas pressure history in a pressurizer tank.

Systems and Subsystems Simulated

Piping networks that include primary coolant systems and secondary systems or other multiloop complex piping networks.

Modeling

1. The unsteady fluid motion is described by the mass, energy, and momentum conservation laws and the fluid equations of state and the initial and boundary conditions for all dependent variables. The well-known water-hammer equations for liquid-filled pipes are evaluated by use of wave superposition. Cases with friction and without friction can be considered, if desired.

2. Piping networks consisting of dead ends, elbows, orifices, multi-branch tees, variable pipe diameters (flow cross-sectional area), check valves, exit valves, and breaks can be considered.

Limitations

1. Allows a maximum of 100 piping legs with 50 nodes per leg, 2 different valves or breaks (additional valves or breaks have characteristics of the second valve or break), 4 pumps with check valves, and 1 pressurizer tank. One case is allowed per execution. More legs and nodes can be configured by changing dimension and data statements.

2. Pump turbine operation range is not considered because it is assumed that each pump is provided with a check valve.

3. Results are valid only for the portion of the system in which the fluid is subcooled and into which no waves from a moving two-phase boundary have penetrated.

Computer Operations

1. Operable on the IBM 360 and 370 using OS/360 and OS/370, respectively, and on the CDC 6600/7600 using Scope 2.1.3. Written in FORTRAN IV and BAL (for the IBM 360) and FORTRAN IV (for the CDC 7600).

2. Requires 480,000 bytes of memory for the IBM 370/195 and 101,000 (octal) words for the CDC 7600.

3. Run time for a typical problem is approximately 3 minutes for the IBM 370/195 and approximately 40 seconds for the CDC 7600.

Interface/Overlap

WATER-HAMMER, WHAM, and WHAM 2 are earlier versions.

BURST-1 (p. A-11) performs similar calculations.

References

Stanislaw Fabic, Computer Program WHAM for Calculation of Pressure, Velocity and Force Transients in Liquid Filled Piping Networks, 67-49-R, November 1967.

V. T. Berta, WHAM/MOD-006 Summary, Aerojet Nuclear Company, December 11, 1975.

NSIC Programming Note 76-25, March 3, 1976, revised July 11, 1977.

P. G. Prassinos, Quick Look Report for LOFT Nuclear Loss-of-Coolant Experiment L2-2, QLR L2-2, June 1978.

[redacted]

Contacts

V. T. Berta
EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, Idaho 83401

H. Edwards
NSIC
Argonne National Laboratory
9700 South Cass Avenue
Argonne, Illinois 60439

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WRAP-EM

Purpose

WRAP-EM is a modular system of computer codes that provides the computational tools required to perform a complete analysis of postulated LOCAs in LWRs. Both PWRs and BWRs can be modeled for large- or small-break LOCAs.

Status

A.	Development	Developed and under improvement by the Savannah River Laboratory, Aiken, South Carolina.
B.	Availability	Operational and under improvement.
C.	Utilization	Developed for use primarily by the NRC in interpreting and evaluating reactor vendor evaluation model methods and computed results.

Problems Solved

Determines thermal hydraulic behavior resulting from a LOCA. Calculates fluid parameters and fuel thermal behavior.

Systems and Subsystems Simulated

Complete reactor system including all components and core.

Modeling

1. The WRAP-EM package is based on a modified version of RELAP4; the included codes are structured around this code format.

2. The package consists of the following code or subcode modules:

a.	WRAPEX	directs the sequences in which the other modules are executed.
b.	WRAPIT	reads input records and creates the WRAP intermediate data set (WIDS) through which other modules communicate.
c.	WRIN	initializes the thermal-hydraulic state describing the reactor system.
d.	TWRAM	calculates the transient response of the system during the blowdown phase of the LOCA.

e.	WROP	processes the edit tape and creates microfiche plots of system variables versus time.
f.	WRROT	allows the user to restart and continue an earlier calculation from data stored on the restart tape.
g.	MWRROT	allows the user to make modifications in system and problem specifications of an earlier calculation and then restart and continue that calculation.
h.	GAPCON	calculates the fuel gap conductance and stored energy and provides initial fuel conditions used by MOXY, WRIN, and TWRAM.
i.	MOXY	calculates the fuel hot plane temperature using transient response data from TWRAM and NORCOOL.
j.	NORCOOL	calculates the transient response of the system during the reflood portion of the accident.

Limitations

1. The limitations of each subcode or code module are incorporated into the package with some modifications for data transfer and the interface between modules.
2. The package is based on the evaluation model assumptions of 10 CFR 50 and gives conservation results.

Computer Operations

1. Operable on the IBM computer under the JOSHUA operating system at the Savannah River Laboratory. Written in FORTRAN IV.
2. Developed for interactive utilization. A modified JOSHUA simulator is available for card input.
3. Employs dynamic dimensioning.
4. A CDC version is being developed.
5. Typical computer run time is dependent upon problem specifications.

References

M. M. Anderson, WRAP - A Water Reactor Analysis Package, DPST-NUREG-77-1 (Aiken, South Carolina: Savannah River Laboratory, E. I. du Pont de Nemours and Company, June 1977).

M. V. Gregory, User's Guide to Input for WRAP - A Water Reactor Analysis Package, DPST-NUREG-77-2 (Aiken, South Carolina: Savannah River Laboratory, E. I. du Pont de Nemours and Company, June 1977).

D. A. Sharp, The PWR Steady-State Capability of WRAP - A Water Reactor Analysis Package, DPST-NUREG-77-3 (Aiken, South Carolina: Savannah River Laboratory, E. I. du Pont de Nemours and Company, April 1975).

D. A. Sharp, The BWR Steady-State Capability of the WRAP-EM System, DPST-NUREG-78-1 (Aiken, South Carolina: Savannah River Laboratory, E. I. du Pont de Nemours and Company, December 1978).

M. R. Buckner et al., The BWR Loss-of-Coolant Accident Capability of the WRAP-EM System, DPST-NUREG-78-2 (Aiken, South Carolina: Savannah River Laboratory, E. I. du Pont de Nemours and Company, December 1978).

H. C. Honeck, The JOSHUA System, ERDA Report DP-1380 (Aiken, South Carolina: Savannah River Laboratory, E. I. du Pont de Nemours and Company, 1975).

C. E. Beyer et al., GAPCON-THERMAL-2: A Computer Program for Calculating the Thermal Behavior of an Oxide Fuel Rod, BNWL-1897 and BNWL-1898 (Richland, Washington: Battelle Pacific Northwest Laboratories, November 1975).

D. R. Evans, The MOXY Digital Computer Program for Boiling Water Reactor Core Thermal Analysis, RE-A-77-081 (Idaho Falls: Idaho National Engineering Laboratory, September 7, 1977).

J. G. M. Anderson et al., NORCOOL-1, A Model for Analysis of a BWR Under LOCA Conditions, A Revised Report, NORHAV-D-47 (Denmark: RISO National Laboratory, August 1977).

RELAP4/MOD5 - A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems - User's Manual, ANCR-NUREG-1335, (Idaho Falls: Idaho National Engineering Laboratory, Aerojet Nuclear Corporation, 1976).

WREM-TOODEE2/MOD3

Purpose

WREM-TOODEE2/MOD3, which is part of the Water Reactor Evaluation Model (WREM) package, performs thermal analysis of a nuclear fuel element.

Status

A.	Development	Developed by the NRC from the TOODEE2 code.
B.	Availability	Released to the National Energy Software Center (NESC) in 1976. Cost $82.00.
C.	Utilization	Used for audit calculation on reactor accident consequences and evaluated fuel performance under accident or post-accident conditions.

Problems Solved

1. Evaluates fuel element thermal response during PWR postulated accidents.
2. Calculates temperature profile in an LWR fuel element.
3. Capable of calculating clad deformation and rupture.

Systems and Subsystems Simulated

Nuclear fuel element.

Modeling

1. Employs a two-dimensional mesh region defined by orthogonal grid lines in slab or cylindrical geometries (cartesian, cylindrical, or polar).
2. Convective and boiling heat transfer mechanisms are assumed at the boundaries.
3. Solves numerically the two-dimensional, time-dependent heat conduction equation within a mesh region.

Limitations

1. Only axisymmetric geometries are considered.

2. Conduction and radiation are the only heat transfer mechanisms within the boundaries of the mesh region.

Computer Operations

1. Operable on the IBM 360 and 370 using OS/360 and OS/370, respectively. Written in FORTRAN IV.

2. Typical program run time is about 6 to 7 minutes on the IBM 370/195.

Interface/Overlap

The WREM-TOODEE2 code is a modification of the TOODEE2 code.

References

WREM: Water Reactor Evaluation Model, Revision 1, NUREG-75/056, May 1975.

G. N. Lauben, TOODEE2: A Two-Dimensional Time Dependent Fuel Element Thermal Analysis Program, NUREG-75/057, May 1975.

[redacted]

c. Various slip velocity ratio correlations,
d. Various heat transfer coefficient correlations including forced convection, nucleate boiling, transition boiling, film boiling, pool film boiling, critical heat flux, quench and reflood,
e. Clad embrittlement, and
f. Fuel rod bulging and spacer grid effects.

4. Requires the following parameters as input:

a. Geometry and nodal arrangement,
b. Axial power factors,
c. Normalized power versus time,
d. Fluid pressure versus time,
e. Fluid inlet mass velocity versus time,
f. Fluid inlet enthalpy versus time,
g. Radial power profile,
h. Initial gas gap thermal conductivity,
i. Initial cold pellet-to-clad radial gap dimension, and
j. Cracking contributions to fuel pellet radius.

Limitations

1. Allows a maximum of 20 axial levels, which can be evenly or unevenly spaced. The nodes at the hot spot are required to be 3 inches in length.

2. Allows a maximum of 23 radial fuel nodes.

3. Used for evaluation calculations and may not represent best-estimate results.

4. The inputs required should be obtained from other various reactor blowdown and reflood codes.

Computer Operations

Information not readily available.

Interface/Overlap

XTHETA is a modification of the CINDA code with portions of the code being extracted from the MOXY code (p. A-65). BUGLEX is a subroutine of the XTHETA code.

References

F. D. Lang, L. H. Steves, and T. A. Bjornard, XTHETA: Multi-Node Heatup Code for Single Channel Transient Analysis, XN-74-21, Rev. 1, January 1975.

[redacted]

XTHETA

Purpose

XTHETA was developed for predictive and post-test analysis of the thermal response of a fuel rod during all phases of a LOCA transient.

Status

A. Development	Developed by Exxon Nuclear Company, Inc., Richland, Washington.
B. Availability	Proprietary.
C. Utilization	Used to analyze the response of a single fuel rod to a LOCA in accordance with the emergency core cooling (ECC) acceptance criteria of 10 CFR 50 Appendix K. Code calculations have been compared to FLECHT results.

Problems Solved

1. Describes the fluid properties, fuel rod temperature distribution, and the departure from nucleate boiling ratio (DNBR) margins.
2. Calculates the heat transfer coefficient, the metal-water reaction, and fuel rod ballooning or failure.

Systems and Subsystems Simulated

A single fuel rod with its associated fluid channel.

Modeling

1. The fuel rod and channel are modeled by radial and axial increments.
2. Solves the one-dimensional transient conduction equation for a single fuel rod and the one-dimensional transient conservation of energy equation for the coolant in the surrounding channel. Both solutions are linked together through the heat transfer coefficient.
3. XTHETA models
    a. Nonuniform and uniform axial power profiles,
    b. Metal-water reaction, both on cladding exterior and cladding interior following a rupture,

F. F. Cadek et al., PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Fluid Report, WCAP-7665, April 1971.

F. F. Cadek et al., PWR FLECHT Final Report Supplement, WCAP-7931, October 1972.

J. O. Cermak et al., PWR Full Length Emergency Cooling Heat Transfer (FLECHT) Group I Test Report, WCAP-7435, January 1970.

F. F. Cadek et al., PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Group II Test Report, WCAP-7544, September 1970.

[redacted]

ZOCO V

Purpose

ZOCO V determines the time- and space-dependent behavior in containments of water-cooled nuclear power reactors following a LOCA caused by a rupture in a main recirculation or steam pipe.

Status

A.	Development	Developed by the Laboratorium für Reaktorregelung und Anlegensicherung, Technische Universität, München, West Germany.
B.	Availability	Operational.
C.	Utilization	Predicts containment response to a LOCA situation. Used to determine the influence of different parameters on the response of containment systems; can also be used to calculate the flow sizes and shapes to fulfill certain criteria.

Problems Solved

Determines the time- and space-dependent pressure, temperature, and mass flows within a containment along with heat transfer to the containment boundaries. Provisions exist for

a. Simulation of the pressure nodes conditions,
b. Simulation of the pressure vessel blowdown behavior,
c. Calculation of the orifice flow,
d. Calculation of quasi-stationary and nonstationary vent pipe flow,
e. Calculation of variable flow areas,
f. Calculation of space and time temperature distributions in solid structures, stationary and nonstationary,
g. Calculation of heat transfer coefficients, and
h. Calculation of initial values for the differential equations.

Systems and Subsystems Simulated

Water cooled reactor containment.

Modeling

1. Simulates the time- and space-dependent pressure distribution in full pressure containment (PWR-type dry containments) as well as pressure suppression systems (BWR-type wet containments).

2. The number of pressure nodes and the connections between them can be arbitarily chosen.

3. Allows the simulation of different geometric situations for the containment because it is based on several different components.

4. Time-dependent variable flow areas controlled by pressure differences of different types can be simulated.

5. Three different possibilities for the mathematical simulation of a pressure node can be used, whereby thermodynamic equilibrium between phases can be assumed as well as deviations from it.

6. Performs calculation of mass flow rates of compressible one- and two-phase two-component flows with different shaped flow paths. A combined integration and interative procedure is used with a convergence criterion to control numerical precision.

7. The solution of the system of differential and algebraic equations is performed simultaneously for a time-step to obtain nonstationary pressure distributions.

8. The time-step size (program-controlled) is variable so that it is possible to use an optimum time-step to minimize computer run time.

Limitations

1. The number of nodes and the complexity of the connections between nodes are limited only by the capacity of the digital computer being used.

2. Requires separate short- and long-term calculations to fully describe the transient.

Computer Operations

1. Developed for the Siemens 4004/55 digital computer. Written in ALGOL. However, a FORTRAN IV version is in preparation.

2. The amount of computer memory needed is dependent upon problem size and complexity.

3. Run time is dependent upon problem size and complexity.

Interface/Overlap

ZOCO I and II are earlier versions of ZOCO V.

[redacted]

References

D. Brosche, "ZOCO V, A Computer Code for the Calculation of Time- and Space-Dependent Pressure Distributions in Reactor Containments," Nuclear Engineering and Design, 23 (1972), pp. 239-272.

D. Brosche, Calculations of Time and Space Dependent Pressure Distributions in Containments of Water Cooled Reactors, Proceedings of the Conference on Mathematical Models and Computational Techniques for Analysis of Nuclear Systems, CONF-730414-P1, VC-32, 1973.

6 KILER

Purpose

6 KILER was developed to analyze core heat transfer during a LOCA for a General Electric (GE) BWR/6 type nuclear reactor plant.

Status

A.	Development	Developed by Energy Incorporated, Idaho Falls, Idaho.
B.	Availability	Released to the National Energy Software Center (NESC) in 1974. Cost $61.00.
C.	Utilization	Used for predictive and post-test LOCA analysis of core heat transfer for a GE BWR/6 nuclear power plant.

Problems Solved

Heat conduction in the fuel rods, time-dependent decay heat sources, and zirconium-steam reaction rates are calculated.

Systems and Subsystems Simulated

GE BWR/6 core.

Modeling

1. 6 KILER was specially developed for the analysis of heat transfer in the GE BWR/6 type plants.
2. A pseudo-two-dimensional model in which radiation and conduction in the axial direction are assumed to be negligible is used.
3. Treats interchange of thermal radiation among surfaces using the network method. 6 KILER generates its own view factor matrix internal to the code based on an 8 by 8 array of fuel rods, modifying the radiation heat transfer matrix each time an emissivity change occurs on fuel rods, water rod, or canister.
4. Heat conduction in the fuel rods is computed by numerical integration of the one-dimensional Fourier heat conduction equation, assuming the temperature is independent of axial or angular position and the heat generation in the heated regions is independent of position and temperature. The one-dimensional heat conduction solution technique is the same used by the MOXY code. Convection heat transfer coefficients are supplied by the user.

5. Rod No. 37, which is designated the "water rod," is considered a passive element.

6. The time-dependent decay heat sources are calculated by linear interpolation from a table of values.

7. The zirconium-steam reaction rate follows the parabolic rate law.

8. 6 KILER is composed of modular subroutines, each with a distinct function, such as computing radiant heat transfer. Such subroutines may be adapted for solving a wider range of problems.

Limitations

Applies only to the geometry and material of the GE BWR/6 reactor core.

Computer Operations

1. Operational on the IBM 360 and 370 using OS/360 and OS/370, respectively. Written in FORTRAN IV and BAL.

2. Requires 240,000 bytes to execute.

3. Typical run time is about 2 minutes.

Interface/Overlap

MOXY/MOD 1 (p. A-65) is a predecessor of 6 KILER.

Reference

R. W. Griebe, K. D. Richert, M. E. Radd, and C. M. Moser, 6 KILER, VER 001, A Digital Computer Code for BWR/6 Core Heat Transfer Analysis, EI 74-5, February 1974.

APPENDIX B

Nuclear Power Plant Transient Codes

This section contains the computer codes that have been developed and are used to perform analysis of nuclear reactors and reactor components during transients which do not involve LOCAs from the primary coolant system. However, analysis of transients that involve LOCAs from the secondary system is included. These transients include anticipated transients with and without scrams and operational transients.

AIROS2A

Purpose

AIROS2A solves the space-independent reactor kinetics equations, provides for the determination of reactivity by solving the discretized equations that represent the spatial heat and mass transfer model for several fuel channels, and calculates transients due to control rod motion.

Status

A.	Development	Developed at Atomics International, Canoga Park, California.
B.	Availability	Released to the National Energy Software Center (NESC) in 1968. Cost $68.00.
C.	Utilization	Used to simulate reactor core dynamics to sudden reactivity changes as a function of feedback variable (control rod transients, reactivity intiated accident).

Problems Solved

1. Calculates the power, reactivity, inverse reactor period, pressure and thermal conditions.
2. Provides for the monitoring of 90 feedback variables.

Systems and Subsystems Simulated

Reactor core and associated flow channels.

Modeling

1. Variations of the film coefficient with flow are accounted for along with the provision for flow decay, afterglow heating, and fuel-clad contact resistance in the calculation of the overall heat transfer coefficient.
2. Scans can be initiated by delay signals from instruments that sense any calculated quantity (i.e., pressure, inverse period, etc.)
3. Generalized feedback equations are used to provide flexibility in models that represent multichannel heat transfer including conduction and convection, energy, pressure, and other phenomena, e.g., fuel melting, coolant boiling, and voiding burn-out.

4. The reactivity equation is generalized and reactivity feedback variables can be constant or vary with the square-root or reciprocal of temperature. Any feedback variable can initiate a reactivity scan, each with a unique delay time.

5. An input generator computes the conduction and convection coefficient for an N x N nodal, multichannel system using built-in tables of specific heat, density, conductivity, and viscosity for the common fuel, structure, and coolant materials and performs an initial temperature calculation.

6. Conduction, convection, and film coefficients may be input or calculated using the Lyons equation or the Dittus-Boelter equation.

7. A provision to simulate setback or controller action of a single bank of control rods is available.

8. Phenomenological models for change of phase are incorporated.

Limitations

1. The code allows a maximum of 15 delayed neutron precursor groups, 400 feedback variables, and 90 feedback variables printed out.

2. Handles any number of channels and nodes per channel.

Computer Operations

1. Operable on the IBM 360 using OS/360. Written in FORTRAN IV (95%) and 360 Assembly language (5%).

2. Requires 256,000 bytes of IBM 360 memory and an SC-4020 graphical display device.

3. A problem involving 6 delayed neutron precursor groups and 50 feedback variables requires about 1 to 2 minutes of computer run time.

4. A restart feature is provided in which restart cards are punched upon abnormal program termination and/or an input option.

Interface/Overlap

AIROS2A is an updated version of AIROS code.

References

R. A. Blaine and R. F. Berland, Simulation of Reactor Dynamics, Volume I - A Description of AIROS IIA, NAA-SR-12452, September 1967.

R. A. Blaine, AI Environmental Report, May 1967.

R. A. Blaine, Modifications of AIROS IIA, W000048.

AI Computing Notice No. 118, April 4, 1968.

ALMOD

Purpose

ALMOD was developed to cover a wide range of PWR transient and accident simulations for safety analysis, comprising failure of safety system components.

Status

A. Development	In the development and verification stages by Gesellschaft für Reaktorsicherheit (GRS) m.b.h., Federal Republic of Germany.
B. Availability	Proprietary.
C. Utilization	Presently being validated. Has been used to predict start-up transients in 1,300 MWe PWR power plants. Code applications comprise licensing, special safety analysis (i.e., anticipated transients without scram (ATWS), and risk evaluation.

Problems Solved

1. Performs a transient analyses of an average fuel pin and cladding temperature, primary and secondary temperature pressure, and other thermal and hydraulic parameters.
2. Capable of analyzing the following transients:
   - a. Loss of main heat sink with auxiliary power and loss of auxiliary power.
   - b. Maximum steam load increase.
   - c. Maximum feedwater flow reduction due to failure of active component.
   - d. Maximum primary coolant flow reduction.
   - e. Maximum reactivity insertion.
   - f. Primary system pressure relief due to opening of pressurizer valve.

Systems and Subsystems Simulated

Total plant system including primary coolant system, pressurizer, reactor vessel and core, secondary and steam dump systems, and safety and control systems.

Modeling

Modular structure incorporates the following in the main modules:

1. Neutron kinetics using two prompt and six delayed neutron groups solved in one space dimension (axial) using a coarse mesh method.

2. Average fuel pin thermodynamics, simulated with an anodal model of the axial section and radial rings.

3. Primary system fluid dynamics solving the one-dimension homogeneous conservation equations by a finite-difference method using a predictor-corrector scheme.

4. Steam generator and secondary system dynamics using a nonlinear, nodal model including slip (Marchaterre-Hogbend correlations) and two-phase heat transfer (Sieder-Tate, Thom, etc.).

5. Pressurizer dynamics including movable boundaries, thermodynamic nonequilibrium, and flashing.

Limitations

1. Environmental properties do not extend below 400°F in the steam generator.

2. Primary coolant system calculations will terminate if preset (input) pressurizer level and pressure are reached.

3. Steam generator is modeled with time-dependent boundary conditions (feedwater and steam/relief flow) in an open loop fashion, eliminating any closed loop feedback.

4. A single input value for the cladding to coolant heat transfer coefficient limits the accuracy of the solution when the primary coolant system flow rates change during the transient.

Computer Operations

1. Operable on the IBM 360. Written in FORTRAN IV.

2. At present, run time for the basic version (point-kinetics, extended steam generator) is about 2 to 4 times real time on the IBM 360, with computation time optimization not yet terminated.

Interface/Overlap

The UTSG code (p. B-107) is used for the primary to secondary heat transfer calculations.

References

W. Frisch and K-D Schmidt, A PWR Plant Model for the Analysis of Large Amplitude Transients, 3rd Power Plant Dynamics Control and Testing Symposium, Knoxville, Tennessee, September 7 to 9, 1977.

A. Hoeld, "A Theoretical Model for the Calculation of Large Transients in Nuclear Natural-Circulation U-tube Steam Generators (Digital Code UTSG)," Nuclear Engineering and Design, 47 (1978), pp. 1-23.

AXTHRM

Purpose

AXTHRM calculates heat transfer from a triangular lattice of cylindrical pins to coolant in either slug or fully developed laminar flow.

Status

A.	Development	Developed at the Los Alamos National Laboratory, Los Alamos, New Mexico.
B.	Availability	Released to the National Energy Software Center, (NESC) in 1964.
C.	Utilization	Used to study the thermal response of fuel bundles during slug or laminar flow conditions. Can also be used to study heat transfer responses during accidents resulting in these flow conditions.

Problems Solved

Calculates temperature fields and heat fluxes during slug or laminar flow through fuel bundles.

Systems and Subsystems Simulated

Cylindrical fuel rods in bundles with a triangular lattice.

Modeling

1. The solution is analytic and closed except for satisfying boundary conditions at a finite (but arbitrary) number of boundary points.
2. The fuel pin is divided into a nuclear fuel region, the tube wall, and the coolant region.
3. Temperature and velocity fields occurring in each region are obtained by the method of finite Fourier cosine transformers in the form of finite series representations.
4. Uses assumptions of thermally isotropic materials and constant physical properties.
5. The solution series is truncated at the number of points at which the homogeneous Neumann boundary conditions are satisfied. The expansion coefficients in the series are obtained by matrix inversion.

[redacted]

Limitations

1. An infinite perfect lattice of fuel elements is considered.
2. No allowance for gap conductance and physical property changes during a transient may limit the accuracy of the results.

Computer Operations

1. Operable on the IBM 7094 using the FORTRAN II monitor with TAPE10 as input and TAPE9 as output.
2. Requires 16,000 bytes of storage.
3. Run time is about 1 to 3 seconds per set of input parameters.

References

R. A. Axford, Multiregion Analysis of Temperature Fields and Heat Fluxes in Tube Bundles with Internal, Solid Nuclear Heat Sources, LA-3167, September 8, 1964.

W. H. Hannum, "Input for Codes for Calculating Thermal Properties of Tubes in HEX Lattices," Memorandum, June 15, 1965.

[redacted]

BEHAVE

Purpose

BEHAVE performs analysis of the thermal-mechanical performance of fuel-pins under steady-state and transient overpower conditions up to and including clad failure.

Status

A. Development	Developed and under improvement at the General Electric Fast Breeder Reactor Division (GE-FBRD). Current version is BEHAVE-SST.
B. Availability	BEHAVE-SST is operational at GE-FBRD and Lawrence Livermore Laboratories. BEHAVE-3 is an earlier version of BEHAVE-SST and BEHAVE-2 (restricted use). Released to the National Energy Software Center (NESC) in 1972. Cost $85.00 (CDC), $70.00 (Honeywell).
C. Utilization	Used by GE-FBRD and the Hanford Engineering Development Laboratory (HEDL) to design and predict overpower experiments and to confirm MELT-III predictions of fuel-rod failure time and location. Primarily used for fast reactor analysis but can be adapted to LWRs.

Problems Solved

1. Determines steady-state and transient fuel and cladding temperature distribution in two dimensions (R-Z).
2. Determines steady-state cladding stress and deformation fields in two dimensions (R-Z).
3. Determines steady-state fuel restructuring, fission gas production, and fission gas release.
4. Determines transient cladding differential thermal expansion and loading, fuel clad friction (or slippage), pressurization of melt, and pressurization due to fission gas release.
5. Cladding failure is also predicted.
6. BEHAVE-2 is the steady-state version incorporated into BEHAVE-SST.

Systems and Subsystems Simulated

Nuclear fuel pin performance.

Modeling

1. Fuel is divided into structural (outer), transport (inner), and nontransport (middle) zones.

2. Radial and transverse full cracking and axial motion of molten fuel are considered.

3. Fuel structure is determined during steady-state operation by a combination of swelling, sintering, cracking, elastic and plastic creep, hot pressing, and melting.

4. Two-dimensional stress-strain calculations are performed by finite-element analysis.

5. Transient fission gas is released at the fuel solids temperature unless the user specifies release fraction at some lower temperature.

6. During steady-state operation, the fuel matrix material (fuel volume exclusion of crack volume and volume of large pores) swells at an inexorable rate with burn-up.

7. After the molten fuel volume fills the central void of an axial section and fuel porosity has been removed by hot pressing, excess fuel is moved to the nearest axial section with available space. Fuel motion options are available, allowing fuel to flow into axial blankets and gas plenum regions.

8. Fuel thermal expansion is described by an empirical correlation, which is a function of fuel stoichiometry and plutonium fraction. Oxygen build-up with burn-up is included but redistribution is not.

9. Cladding failure is predicted using a LIFE fraction type cumulative damage function based on the Larson-Miller parameter correlations.

Limitations

1. Allows a maximum of 5 axial sections, 10 radial cladding rings, 20 radial fuel rings (for transients), and 30 fuel rings for steady-state analyses.

2. Cladding mechanical properties are for irradiated 20% CW type 316 stainless steel. Annealed type 316 properties are available but the model is less detailed.

3. Calculates only single rod analysis with no interaction between rods.

4. Treatment of the following phenomena is not considered:

   a. Fission-gas induced fuel swelling,
   b. Strain-based cladding failure models,
   c. Fuel vapor pressure loading,
   d. Local (circumferential) thermal or mechanical effects,
   e. Oxygen redistribution models,
   f. Cladding wasting models, and
   g. Coolant boiling.

Computer Operations

1. Operational on the Honeywell-6000 and the CDC 7600. Written in FORTRAN IV.

2. Batch mode with extensive overlay on both computers is employed.

3. Requires 70,000 bytes of memory on the Honeywell 6000 and 150,000 bytes of small-core memory on the CDC 7600 (MNF compiler). Disc storage is required for input data not internally calculated.

4. Typical run time for the CDC 7600 with the maximum node structure for a steady-state and 1-second transient with internally calculated temperatures requires 48 seconds. When temperatures are supplied, a 15-second transient requires 287 seconds, and, when a 3-second transient is run with 30 transient times, the program requires about 375 seconds. Temperature calculations require about 50% of the calculation time, with temperatures supplied on cards.

5. The cladding failure time and location predictions have been obtained using the BEHAVE-SST code for 3 unirradiated and 11 irradiated TREAT overpower experiments.

Future Efforts

Continued validation with experimental data and treatment of some of the code limitations is being performed.

References

S. Oldberg, BEHAVE-2, Oxide Fuel Performance Code in Two Spatial Dimensions and Time, GEAP-13788, February 1972.

S. Oldberg and R. G. Stuart, Steady State and Transient Fuel Mechanism: The BEHAVE-3 Code, GEAP-14021, August 1974.

R. R. Sherry and D. B. Atcheson, User's Manual for the BEHAVE-SST (Steady State and Transient), Fuel Rod Mechanics Program, GEFR-00001, February 1977.

D. B. Atcheson, R. R. Sherry and K. J. Shimane, "Cladding Failure in TREAT Overpower Experiments: A Mechanistic Interpretation and Its Implications for LMFBE Safety Analysis," CONF-7600lm, ANS International Meeting on Fast Reactor Safety and Related Physics, October 1976.

"Safety Engineering," Ninth Quarterly Report for August-October 1974, GEAP-13923-9, GE-FBRD, November 1974.

"Safety Engineering," Tenth Quarterly Report for November-December 1974 and January 1975, GEAP-13923-10, GE-FBRD, February 1975.

"Safety Engineering," Eleventh Quarterly Report for February-April 1975, GEAP-13923-11, GE-FBRD, June 1975.

"Safety Engineering," Twelfth Quarterly Report for May-July 1975, GEAP-13923-12, GE-FBRD, August 1975.

"Safety Engineering," Thirteenth Quarterly Report for August-October 1975, GEAP-13923-13, GE-FBRD, November 1975.

"Safety Engineering," Fifteenth Quarterly Report for February-April, GEAP-13923-15, GE-FBRD, May 1976.

"Safety Engineering," Sixteenth Quarterly Report for May-July 1976, GEAP-13923-16, GE-FBRD, August 1976.

"Safety Engineering," Seventeenth Quarterly Report for August-October 1976, GEAP-13923-17, GE-FBRD, November 1976.

"Fuel Rod Mechanics and Failure Analysis," Eighteenth Quarterly Report for November-December 1976 and January 1977, GEFR-13923-18, GE-FBRD, February 1977.

"Fuel Rod Mechanics and Failure Analysis," Nineteenth Quarterly Report for February-April 1977, GEFR-13923-19, GE-FBRD, May 1977.

"Fuel Rod Mechanics and Failure Analysis," Twentieth Quarterly Report for May-July 1977, GEFR-13923-20, GE-FBRD, to be published.

BOIL1

Purpose

BOIL1 calculates the core heat-up and meltdown sequences in coolant boil-off accidents.

Status

A. Development	Developed by Battelle Memorial Institute, Columbus Laboratories, Columbus, Ohio.
B. Availability	Released to the National Energy Software Center (NESC) in 1977. Cost $57.00.
C. Utilization	Used in the Reactor Safety Study (WASH-1400) and used to analyze the TMI incident.

Problems Solved

Calculates heat produced by fission product decay, water-metal reaction and thickness, core mixture level, coolant boil-off rate, and core meltdown.

Systems and Subsystems Simulated

Nuclear core and fuel.

Modeling

1. The core is divided into small volumes or nodes for a time-step (delta t) calculation.
2. Explicit finite-difference approximation is used to solve the two-dimensional equations.
3. Small core coolant makeup may be used.

Limitations

1. Allows a maximum of 50 axial and 10 radial nodes.
2. The coolant heat transfer model assumes a relatively slow transient in which the heat transfer is not conduction limited.
3. Considers only constant pressure transients.

Computer Operations

1. Operable on the CDC Cyber 73, 7600 under Scope 3.4.4. Written in FORTRAN IV.
2. Requires 31,000 (octal) words of memory.
3. Typical run time is 40 seconds with a 1-minute time-step size.

Interface/Overlap

NURLOC-1.0 (p. A-79) is the predecessor of BOIL1.

References

R. C. Wooten, BOIL1, A Computer Program to Calculate Core Heatup and Meltdown in a Coolant Boiloff Accident, BCL, 1976.

Reactor Safety Study, An Assessment of Accident Risk in U.S. Commercial Nuclear Power Plants, Appendix VII-A, WASH-1400, August 1974.

CADD

Purpose

CADD determines the transient response of a fuel pin (or pins) and associated coolant channel to various reactor plant transients including anticipated transients without scram (ATWS).

Status

A.	Development	Developed by Babcock and Wilcox (B&W), Nuclear Power Generation Division, Lynchburg, Virginia.
B.	Availability	Proprietary.
C.	Utilization	Used to analyze core thermal behavior during various transient conditions. Has been used to study ATWS behavior of B&W plants in response to NRC licensing and safety criteria.

Problems Solved

1. Solves the time-dependent neutron kinetics equations in conjunction with a thermal-hydraulics solution for an average fuel pin during a reactivity transient.

2. Determines coolant and fuel pin temperatures, heat flux from the pin, flow rate, enthalpy, coolant density, heat transfer coefficient, various reactivity components, volume weighted fuel temperatures, net power transferred from cladding to coolant, and reactor power.

3. Options exist to determine metal-water oxidation depths and pressure drops across the core.

Systems and Subsystems Simulated

Simulates an average fuel pin(s) in the core including fuel pellet, gap, cladding, and associated coolant channel. An option to simulate the reactor coolant loop is available.

Modeling

1. The fuel pin and coolant channel are given detailed spatial representation by axial and radial sectionalization.

2. The loop simulation utilizes a region averaged model.

3. The kinetics model incorporates reactivity feedback effects because of moderator density, temperature changes, and Doppler broadening.

4. The conservation equations of mass and energy in the coolant passage assume homogeneous, one-dimensional flow.

5. The transient situation occurs from a steady-state condition or at a user-supplied rate; transient conditions can be imposed on the system through reactivity insertion (or heat generation) due to control rod insertion or ejection, core inlet enthalpy (or temperature) variations, core inlet flow variations, and system pressure variations.

6. Provision exists for handling three representative fuel pins in the core in a manner similar to that used for the average pin. This allows a description of the temperature distribution in the core or an analysis of several cases simultaneously.

7. The appropriate conditions associated with the heat transfer regions of conduction and convection, nucleate boiling, or film boiling are applied to the surface of the fuel pin.

8. Includes effects from two-phase flow and the zirconium-water reaction using the parabolic rate law.

9. Reactor core evaluations can be made independent of the primary loop by using, as input, reactor inlet enthalpy, reactor inlet flow, system pressure, and either reactivity or heat generation as a function of time. The use of the loop option requires that reactor inlet flow versus time and reactivity versus time be input along with steam generator heat removal.

Limitations

1. A maximum of three pins can be analyzed with 3 to 12 axial sections, 3 to 8 radial sections in the fuel pellet, and 1 to 3 radial sections in the clad.

2. Four volumes are used in the loop model: reactor core, hot leg, steam generator, and cold leg. A pressurizer model is also used which includes the heater, spray, and safety valve simulations.

3. Superheat is considered as an extension of the film boiling region. Neither transition boiling nor axial conduction is considered.

4. Two-phase flow is represented by a fog flow model, and no void formation is considered in either the subcooled or boiling state. This affects the reactivity feedback mechanisms.

5. The loop fluid is always considered to be subcooled.

6. The loop calculation considers only one loop.

Computer Operations

?

Interface/Overlap

CADD is used in conjunction with other B&W programs to analyze complete transient situations. It is a major modification of the CHIC-KIN code (p. B-29).

References

R. H. Stoudt and S. E. Busby, CADD-Computer Application of Direct Digital Simulation of Transient in Water Reactors, BAN-10076, August 1973.

J. A. Redfield, CHIC-KIN - FORTRAN Program for Intermediate and Fast Transients in Water-Moderated Reactors, WAPD-TM-479, January 1965.

[redacted]

CESEC

Purpose

CESEC provides a digital simulation of the nuclear steam supply system and facilitates the analysis of abnormal or accidental operational conditions by means of the study of postulated transients, anticipated transients with/without scram, and operation transients.

Status

A.	Development	Developed by Combustion Engineering Inc., Nuclear Power Division, Windsor, Connecticut.
B.	Availability	Proprietary.
C.	Utilization	Used for the safety analysis of nuclear power reactors for licensing purposes.

Problems Solved

Determines the response of nuclear power reactors and provides the prediction of the thermal and hydraulic conditions of the fluid and structural material for the system.

Systems and Subsystems Simulated

Complete nuclear reactor systems excluding turbines and generators.

Modeling

1. CESEC is a modular code that contains separate subroutines that perform one or more distinct functions. This facilitates modification and updating.

2. Modeled by control volumes and flow paths.

3. Simultaneous numerical integration of the set of nonlinear, first-order differential equations with time-varying coefficient which mathematically describes the physical models of the several subsystems is performed by a predictor-corrector Runge-Kutta scheme.

4. The dynamic functions provided by the code are point kinetics neutron behavior, Doppler and moderator reactivity feedback, boron and control element assembly (CEA) reactivity effects, multinode average and hot channel reactor core thermal-hydraulics, reactor coolant pressurization and mass transport, reactor coolant system safety valves, steam generation, steam generator water level, main steam system bypass, safety and turbine valves, as well as alarm, control, protection, and engineered safety features.

5. Allows for two-phase effects by empirical formulation and correlations.

6. Calculates the following transients:

   a. Loss of load,
   b. Loss of feedwater flow,
   c. Loss of coolant flow,
   d. CEA withdrawal from subcritical, low power, and critical conditions,
   e. CEA misoperation,
   f. Boron dilution,
   g. Idle loop start-up,
   h. Loss of all normal ac power,
   i. Excess heat removal,
   j. Minor secondary system pipe breaks,
   k. Steam generator tube rupture,
   l. Major secondary system pipe breaks, and
   m. Reactor coolant pump shaft seizure.

Limitations

1. The reactor plant is modeled with 2 loops, each having 14 nodes, and the reactor vessel, which has 3 nodes, arranged as follows:

   a. The reactor vessel has an inlet plenum, active core, and outlet plenum,
   b. Two nodes for each hot leg,
   c. Two nodes for each cold leg, and
   d. The steam generator has an inlet plenum, active tube region, and outlet plenum.

2. The pressurizer is modeled with a water and steam volume.

3. The steam generator secondary is modeled with a water and steam volume.

4. For neutronic calculation, 1 energy group and 6 delayed groups are used and an 11-group decay heat model is employed for power calculations.

5. For the fuel pin in the average channel, four radial nodes (three in the fuel rod and one in the coolant subchannel) and up to five thermal axial nodes are used.

Computer Operations

Operable on the Cyber 70/Model 76 CDC 7600 computer system. Written in FORTRAN IV.

Interface/Overlap

The RELAP (p. A-95) and FLASH (p. A-45) codes are predecessors of the CESEC code.

Reference

CESEC - Digital Simulation of a Combustion Engineering Nuclear Steam Supply System, ECNPD-107, Rev. 1, Combustion Engineering Inc., June 1974.

CHART

Purpose

CHART was developed for the thermal analysis of the charcoal adsorber in a reactor containment system.

Status

A. Development	Developed by the Oak Ridge National Laboratory, Oak Ridge, Tennessee.
B. Availability	Released to the National Energy Software Center (NESC) in 1978. Cost $64.00.
C. Utilization	Used for steady-state and transient accident calculation of charcoal adsorbers. The principal consideration is the temperature build-up within the charcoal, which may cause fires.

Problems Solved

Maximum temperatures caused by iodine adsorption under accident conditions are estimated along with other thermal-hydraulic conditions.

Systems and Subsystems Simulated

Charcoal adsorber in reactor containment.

Modeling

1. The charcoal adsorber is divided into regions with lattice points in each region.
2. A differential equation relating slab temperature to heat generation rate, air flow rate, and the heat parameters of air and charcoal is solved in both one and three dimensions.
3. The program uses the heat transfer techniques employed in the HEATING3 code, which is a general heat transfer code for steady-state calculation. The Liebmann method and the AITKEN delta-squared extrapolation process are used to solve the finite difference equations which approximate the partial differential equations. For a transient problem, a first forward time difference technique is used with the solution expressed explicitly.
4. Heat generation is expressed as a function of position, time, and temperature.

5. The cooling gas temperature is calculated as a function of temperature along the flow direction and included in the heat generation term, making it possible for what is basically a heat conduction code to handle combined conduction and convection adsorber problems.

Limitations

Limited to a maximum of

1. 100 regions,
2. 1,750 lattice points,
3. 50 materials,
4. 50 boundary conditions,
5. 100 fine lattice lines along any axis,
6. 50 gross lattice lines along any axis,
7. 20 heat generation functions,
8. 25 initial temperature functions,
9. 50 position-dependent functions, and
10. 20 time-dependent functions.

Computer Operations

1. Operable on the IBM 360, 370, and 3033 computers using OS/360, 370. Written in FORTRAN IV.

2. Requires 465,000 bytes of storage for execution and standard input and output units.

3. The sample problem required less than 15 seconds of run time on the IBM 3033.

Interface/Overlap

HEATING3

References

R. P. Shields and M. Siman-Tov, The Effect of Iodine Decay Heat on Charcoal Adsorbers, ORNL-4602, April 1971.

W. D. Turner and M. Siman-Tov, HEATING3 - An IBM 360 Heat Conduction Program, ORNL-TM-3208, February 1971.

W. D. Turner and J. S. Crowell, Notes on HEATING - An IBM 360 Heat Conduction Program, CTC-INF-980, November 1969.

R. P. Shields, "Ignition of Chemical Adsorbers by Fission Product Decay Heat," Proceedings of the 11th Air Cleaning Conference, Hanford, Washington, August 31-September 3, 1970, pp. 770-790.

CHIC-KIN

Purpose

CHIC-KIN treats fast and intermediate reactivity transients in water-cooled nuclear reactors.

Status

A. Development	Developed by Westinghouse Electric Corporation at the Bettis Atomic Power Laboratory, West Mifflin, Pennsylvania.
B. Availability	Released to the National Energy Software Center (NESC) in 1970. Restricted to use by U.S. citizens for either official U.S. Government or commercial nonmilitary application at U.S. installations. Cost $68.00.
C. Utilization	Used to study reactivity transients in nuclear power plants and for preliminary safety analysis report (PSAR) and final safety analysis report (FSAR) requirements.

Problems Solved

Calculates the power, temperatures, and pressure surges in the reactor vessel when control rod motion, inlet temperature, inlet flow, and system pressure are known functions of time.

Systems and Subsystems Simulated

Nuclear reactor vessel and fuel region.

Modeling

1. The reactor model considered is a single-pass, water-cooled core represented by a single fuel element-coolant passage system with a detailed spatial representation of the fuel element and core by axial and radial sectionalization.
2. Performs a simultaneous solution of the basic one-dimensional conservation equations in each section.
3. Provides calculations of the subcooled boiling (local boiling) void fractions, which are a significant contributor to reactor shutdown for a large class of reactors.

4. Fluid dynamics are represented by a momentum integral model which allows stagnant initial conditions, flow reversal, and internal channel pressure build-up.

5. The model accounts for details of nonboiling, nucleate boiling, and film boiling heat transfer.

6. For heterogeneous reactors, all known major feedback mechanisms are accounted for. Reactivity feedback to the kinetics equations due to moderator density change, moderator temperature change, fuel expansion, and Doppler broadening are provided.

Limitations

1. The code considers a maximum of

    a. 1 through 20 axial sections,
    b. 2 through 10 radial sections,
    c. 2 through 30 mass velocity or pressure drop pairs,
    d. 2 through 30 inlet enthalpy pairs,
    e. 2 through 30 exit enthalpy pairs, and
    f. 2 through 30 system pressure pairs.

2. No provision for molten fuel motion.

3. Requirements for the Bettis Environmental Routine may make operation of this core on other systems impossible without modification.

Computer Operations

Operable on the CDC 6600 using Scope 3.1. Written in FORTRAN IV.

Interface/Overlap

CHIC-KIN uses the Bettis Environmental Routine.

References

J. A. Redfield, CHIC-KIN - A FORTRAN Program for Intermediate and Fast Transients in a Water Moderated Reactor, WAPD-TM-479, January 1965.

C. J. Pfeifer, CDC6600 FORTRAN Programming, Bettis Environmental Report, WAPD-TM-668, January 1967.

COAST

Purpose

COAST determines the piping system response of a two-loop, four-pump pressurized water reactor (PWR) to a LOCA flow transient.

Status

A. Development	Developed by Combustion Engineering, Inc., Nuclear Power Division, Windsor, Connecticut.
B. Availability	Proprietary.
C. Utilization	Used for the design and analysis of the plant response to a LOCA flow transient. The code analysis ensures that the as-built loop hydraulics will not invalidate the safety analysis report (SAR) loss of fluid flow analysis. Calculations have been compared to test data obtained from the Palisades Nuclear Generating Station.

Problems Solved

Calculates piping flows, piping pressure drops, and steam generator pressure drops along with pump speed.

Systems and Subsystems Simulated

A two-loop, four-pump PWR primary coolant system including piping, steam generator primary side, and reactor vessel.

Modeling

1. The reactor coolant system is modeled with seven flow sections and four nodal points. The nodal points are located in the reactor vessel inlet and outlet plenums and in the outlet plenums of the steam generators.

2. The conservation of momentum equation is applied at each node and is also applied, assuming an unsteady, homogeneous, incompressible fluid, for each flow segment.

3. Pump dynamics are modeled by the traditional head-flow characteristic curve for fully operable pumps and with a four-quadrant diagram (parametric curves of pump head and speed versus flow) for performance at other than rated speeds.

4. The governing conservation equations of fluid mass and loop momentum are solved simultaneously with the pump dynamic equations using a finite difference technique with a Newton-Raphson iteration scheme to produce a converged solution within specified criteria at each time-step.

5. Any combination of pump coastdowns can be simulated, and the use of symmetry for the system reduces the number of combinations to 15. These cases can be analyzed with one computer run.

6. Both forward and reverse flow through all components is considered.

Limitations

1. Does not consider a two-phase fluid and its effect on pump operation. Therefore, transient with pressure reduction cannot be simulated.

2. Calculates steady-state before and after a coastdown but will not calculate the transition to natural circulation following the loss of electrical power to all pumps.

3. For two-pump coastdown on opposite loops, COAST cannot distinguish between pumps on the diagonal and pumps on the same side of the reactor vessel since the reactor vessel inlet is represented as one nodal point.

Computer Operations

Operable on the CDC 6600. Written in FORTRAN IV.

Reference

R. S. Daleas, COAST Code Description, Report CENPD98, April 1973.

DSNP

Purpose

DSNP was developed to facilitate the dynamic simulation of a large variety of nuclear power plants under transient conditions.

Status

A.	Development	Developed by Argonne National Laboratory, Argonne, Illinois, under sponsorship by the Department of Energy, Division of Research and Technology.
B.	Availability	Released to the National Energy Software Center (NESC) in 1978. Cost $101.00.
C.	Utilization	Used to simulate transients in nuclear power plants for design and analysis.

Problems Solved

Determines the dynamic behavior of the system and analyzes the transient thermal-hydraulic response of coolant, core, and structural material.

Systems and Subsystems Simulated

Complete nuclear power plants or individual subsystems and components.

Modeling

1. Used interchangeably for a DSNP computer language. DSNP language is a special purpose, block-oriented digital simulation language for which FORTRAN is considered a subset.
2. Physical components are represented by functional blocks or modules in which complex components are represented by several modules.
3. Specific modules are present for all components and operations such as the core hydraulics, kinetics, pipes, pumps, connectors, integrator, auxiliary functions, etc.
4. The mathematical models in each module vary from module to module. However, most modules used a lumped parameter model. The differential equation produced by each module is solved by a centralized integration scheme.

Limitations

1. There is no limitation on the complexity of the problem and the number of modules used.
2. Only overall plant transients can be simulated since most modules are based on a lumped parameter approximation.

Computer Operations

1. Operable on the IBM 370 using OS/370. Written in DSNP and FORTRAN IV.
2. Requires 250,000 bytes of storage, six scratch files requiring disk or magnetic-tape storage, and a permanent library file. The code uses batch input.
3. Program run time is a function of the size of the simulation. The precompilation time is about 50 seconds for 1,000 FORTRAN statements.

Future Efforts

Future modifications to DSNP will increase its capability so that individual components can be analyzed.

References

D. Saphier, The Simulation Language of DSNP, ANL-CT-77-20, Rev. 2, September 1978.

D. Saphier, The Level One Modules Library for DSNP, ANL-CT-77-21, Rev. 2, September 1978.

D. Saphier, The Precompiler for DSNP, ANL-CT-77-22, Rev. 2, September 1978.

D. Saphier, "A Simulation Language for Nuclear Power Plants," Proceedings of the 1977 Summer Computer Simulation Conference, July 18-20, 1977, AFIPS Press, Montvale, New Jersey, 1977, pp. 353-358.

DYNAM

Purpose

DYNAM performs a dynamic analysis of once-through boiling flow oscillations with steam superheat.

Status

A.	Development	Developed by General Atomic Corporation, San Diego, California.
B.	Availability	Released to the National Energy Software Center (NESC) in 1970. Cost $66.00.
C.	Utilization	Used to study flow oscillations in reactor geometries. Can be used with supercritical pressurized water systems.

Problems Solved

Determines system thermal-hydraulic parameters and flow oscillation frequencies.

Systems and Subsystems Simulated

?

Modeling

1. The model describes nonboiling, boiling, and superheat flow oscillations in a nodalized core, steam generator, or other heat transfer arrangements.
2. The superheat regime model is single-phase, variable density for subcritical pressure operation and is also applicable to the once-through operation using supercritical pressurized water.
3. Linearized partial differential conservation equations are solved using a Laplace transformation of the temporal terms and integration of the spatial variations. DYNAM is written in complex variable notation.
4. The solution has no numerical instabilities.

[redacted]

Limitations

DYNAM allows a maximum of

1. 29 boiling nodes,
2. 40 superheat nodes,
3. 30 intervals used to describe the power distribution in the non-boiling and boiling regions,
4. 7 intervals and corresponding friction multipliers read in per case,
5. 14 exit qualities read in per case,
6. 10 coefficients read in for the 20 *vs* K polynomial fit,
7. 48 frequencies at which open loop frequency response is desired, and
8. 48 frequencies at which signal output is desired.

Computer Operations

1. Operable on the Univac 1108 using EXEC II. Written in FORTRAN IV.
2. A typical problem requires 5 seconds of computer run time on the Univac 1108.

Future Efforts

Improvement of the pressure drop equations.

Interface/Overlap

The STABLE code performs similar calculations.

References

L. E. Efferding, *DYNAM, A Digital Computer Program for Study of the Dynamic Stability of Once-Through Boiling Flow with Stream Superheat*, GAMD-8656, June 26, 1968.

E. F. Efferding, *Static and Dynamic Stability of Steam Water Systems, Part 1, Critical Review of the Literature*, GA-5555, October 1964.

[redacted]

FACTRAN

Purpose

FACTRAN determines the transient thermal behavior in a cross-section of a metal-clad UO_2 fuel rod.

Status

A.	Development	Developed by Westinghouse Electric Corporation, Nuclear Energy Systems Division, Pittsburgh, Pennsylvania.
B.	Availability	Proprietary.
C.	Utilization	Used for the design and analysis of nuclear fuel rods and for safety analysis of Westinghouse nuclear power plants in response to NRC criteria. Also used to determine the fuel rod response to anticipated transients without scram (ATWS).

Problems Solved

Determines the transient temperature distribution in the fuel rod along with the metal-water reaction, heat generated, and stored energy in the fuel, heat flux, fuel and clad expansion, clad strain, and coolant temperatures.

Systems and Subsystems Simulated

Nuclear fuel rod and the associated flow channel cross section at a particular axial location.

Modeling

1. The fuel rod is modeled by concentric rings.

2. The following modeling is used:

   a. A sophisticated gas gap heat transfer model,
   b. A parabolic power distribution in the fuel pellet,
   c. A heat transfer calculation, which includes forced subcooled convection (Dittus-Boelter), local (subcooled) boiling (Jens-Lottes), critical heat flux, transition boiling, and film boiling (Bishop-Sandberg-Tong),
   d. A zircaloy-metal reaction, which is assumed to be generated uniformly in the mass of the clad,
   e. Fuel and clad melting, which is assumed to be spread over a 5°F zone instead of taking place at a constant temperature, and

[redacted]

f. Automatic time-step selection, which is a function of the temperature difference between time-steps.

3. The heat conduction equations written for each concentric ring form a system of linear equations and are solved simultaneously.

4. Power and local fluid condition (flow, pressure, and temperature) must be input. The gas gap and surface heat transfer can be input by the user or calculated by the program.

Limitations

1. The fuel rod is allowed a maximum of 20 rings with one for the gas gap, one for the clad, and one for the fluid film.

2. The calculation considers only one axial cross section and does not consider axial conduction.

3. Once departure from nucleate boiling (DNB) occurs, this condition remains for the rest of the transient. Rewetting of the fuel rod is not considered.

4. Changes in fuel rod geometry due to melting are not considered and, if melting is calculated to occur, the result should be interpreted with caution.

5. The change in heat transfer characteristics with build-up of an oxide layer is not considered.

Computer Operations

Operable on the CDC 6600 computer. Written in FORTRAN IV.

Reference

H. G. Hargrove, FACTRAN: A FORTRAN IV Code for Thermal Transients in a UO$_2$ Fuel Rod, WACP-7908, July 1972.

FORE/FORE2

Purpose

FORE/FORE2 calculates the reactor response to a programmed reactivity insertion specified as a series of ramps or power changes.

Status

A.	Development	Developed by the Atomic Power Equipment Department and the Advanced Product Operations Division of the General Electric Company.
B.	Availability	Released to the National Energy Software Center (NESC) in 1965. Cost $64.00 (Philco), $79.00 (GE-635), $77.00 (IBM).
C.	Utilization	Used to analyze steady-state and transient reactor performance with respect both to design and to accident analysis. Basically developed for use with liquid-metal reactors but can be used for any single-phase coolant.

Problems Solved

1. Calculates the reactor power, the individual reactivity feedbacks, and the temperature of coolant, cladding, fuel, structure, and additional materials.

2. Temperature profiles are computed at specified axial positions for an average channel and the peak power (central) channel.

Systems and Subsystems Simulated

Reactor vessel, core, and nuclear fuel rods.

Modeling

1. The reactor is modeled with axial and radial zones, and the differential equations describing fluid flow, heat transfer, and point neutronics are solved by an explicit finite-difference procedure.

2. Heat of fusion accompanying fuel melting is taken into account.

3. Feedback reactivity mechanisms respond to changes in temperatures, including fuel Doppler effect, core thermal expansion, and axial black or reflector materials.

4. Liquid metal or coolant voiding reactivity and spatial and time variations of the fuel-cladding gap coefficient due to changes in gap size are considered.

5. Separation of the fuel power into prompt and total (including decay heat) is performed.

6. Fuel bowing and core deformation due to axial pressure differences are considered.

7. Has the ability to correctly handle the variation of coolant flow rate in channels with orificing during a coastdown accident.

8. Input consists of property data, geometry, power, flow distribution factors, external time-varying functions, and experimental coefficient and termination data.

Limitations

1. The calculation is performed for up to seven axial positions in three channel types that represent radial zones in the reactor.

2. A maximum of six delayed neutron groups is considered.

3. Reactor excursions which can be calculated are restricted to those transients in which the reactor is not substantially destroyed. As a rule, changes in reactor geometry and composition during an excursion are limited to those cases in which the reactivity effects of the changes may be considered as small perturbations of the initial system; thus, accidents involving large-scale disassembly and bulk meltdown of a core are not covered, and calculations are valid only while the core retains its initial geometry.

Computer Operations

1. Operable on the Philco-2000, GE-635 and IBM 360 computers using SMART (the GE-aped monitor), Gecos (635), and OS/360 (IBM 360), respectively. Written in FORTRAN IV.

2. FORE requires 32,000 bytes of memory and FORE2 requires 40,000 bytes of memory and three files on the GE-635 computer.

3. Run time for FORE is proportional to the number of time-steps, generally 2 to 26 minutes. FORE2 requires from 0.5 to 3.5 seconds per time-step of 1 to 10 μs.

4. Unlimited wrap-up and restart capability with changes in input before restart is provided.

5. The maximum stable time-step is determined by the code.

6. The FORE program consists of the main program, FORE, which handles bookkeeping and error checking, COPH, FEEB, TERM, and OUTPUT. FORE2 consists of three independent codes:

   a. EEE, which performs all calculations and provides a condensed, or short, edit of the results,
   b. FEDIT, which provides a detailed, or long, edit of the results, and
   c. FRCURV, which performs curve plotting of the output.

7. FORE requires as input a minimum and maximum step size, which will place a limit on the doubling or halving process, and an initial step size with which to begin the calculation. FORE2 calls several system subroutines that are not included with the program; however, write-ups of these subroutines are included with the code package.

References

P. Greebler, D. B. Sherer, N. H. Walton, FORE, A Computational Program for the Analysis of Fast Reactor Excursions, GEAP-4090, October 16, 1962, Plus Errata and Addenda, EA2, February 10, 1965, EA3, June 22, 1965, and EA4, December 22, 1965.

J. N. Fox, B. E. Lawler, and H. R. Butz, FORE II, A Computational Program for the Analysis of Steady-State and Transient Reactor Performance, GEAP-5273, September 1966.

Nels J. Anderson, Jr., IDFTAP, A FORTRAN IV Function Subprogram, S6T001, January 15, 1965.
BITA, L5T001, December 3, 1964.

Nels J. Anderson, Jr., FORTRAN IV Subroutine S4T001, January 15, 1965.
FLEXO, J3T001, January 21, 1965.
FLTBCD, L2T006.
LISTFORM, S6T002, February 12, 1964.
NFXBCD, L2T004, November 16, 1965.

C. T. Ehlers, NSORT, P5T001, October 29, 1964.
John W. Weil, PLOTRC, K5T003.
Walter A. Duncan, VAMPIT, K5F009, April 25, 1966.
A. E. Scadden, Zero Division Overflow Modifier.

FRANCESCA BWR

Purpose

FRANCESCA BWR performs analysis of the fluid hydraulic and fuel rod thermal behavior of the core during steady-state or transient conditions for BWR plants. The coolant flow redistribution among different groups of channels is evaluated.

Status

A.	Development	Developed by the Joint Nuclear Research Centre, Ispra Establishment, Italy.
B.	Availability	Operational.
C.	Utilization	Used as a design tool and also to determine fluid and fuel rod transient behavior under input conditions.

Problems Solved

Calculates the flow distribution, fuel centerline and surface temperature, cladding inside and outside surface temperature, coolant temperature, heat flux, void fraction, and vapor quality. The average core pressure drop can also be calculated.

Systems and Subsystems Simulated

Groups of fuel rods and associated coolant channels. Also individual fuel rod internals.

Modeling

1. The core is divided into groups of fuel rods which have an axial mesh.

2. Requires axial heat generation, inlet mass, and energy and physical dimensions as input.

3. The kinetics model is essentially that of Zuber and Findlay and includes the average effect of vapor drift, as well as velocity and concentration distributions.

4. Radial conduction in cylindrical fuel rods is considered. Heat transfer from cladding surface to coolant is governed by the Colburn correlation in convection heat transfer and the Jens-Lottes correlation for boiling heat transfer.

5. Subcooled boiling is accounted for and vapor production in the subcooled region is evaluated by Browring's method.

6. The two-phase flow model considers vapor recondensation in the subcooled region.

Limitations

1. A maximum of 10 groups of fuel rods may be considered with a maximum of 30 axial increments.

2. The heat conduction in the axial direction is ignored and is not significant for the range of problems considered.

3. A departure from nucleate boiling (DNB) model is not employed, and calculations of the evolution of a transient after DNB are not considered; therefore, no film boiling or steam heat transfer is evaluated.

Computer Operations

1. Operable on the IBM 360/65 computer using OS/360,65. Written in FORTRAN and assembled in FORTRAN H level Z.

2. Requires 17,448 hexadecimal bytes for the total program.

3. Estimated run time is roughly 0.1 to 0.2 minute per axial node per 1,000 time-steps on the IBM 360/65.

Interface/Overlap

The FRANCESCA BWR code is part of the CONSTANZA code package.

References

G. Forti, FRANSCESCA BWR - A Numerical Programme for the Steady-State and Dynamic Calculations of Parallel Coolant Channels for BWR Nuclear Reactors, EUR-4902e, 1972.

G. Forti, FRANCESCA, A Dynamic Program for Boiling Cooling Channels, EUR-4241, 1969.

G. Forti, COSTAX BWR, A Numerical Programme for the Axial Dynamics of BWR Nuclear Reactors, EUR-4837, 1972.

G. Forti, A Dynamic Model for the Cooling Channels of a Boiling Nuclear Reactor, EUR-4052, 1968.

GAPCON-THERMAL-2

Purpose

GAPCON-THERMAL-2 was developed as a tool for estimating fuel-gap gas conductance and fuel stored energy.

Status

A. Development	Developed by the Pacific Northwest Laboratory, Richland, Washington.
B. Availability	Released to the National Energy Software Center (NESC) in 1975. Cost $74.00.
C. Utilization	Used to obtain properties for inputs into other codes. Also has been used to study detailed fuel-rod heat transfer.

Problems Solved

Calculates fuel rod power history, fission gas generation and release, fuel relocation and densification, fuel cladding gap conductance, fuel rod stored energy, and burn-up.

Systems and Subsystems Simulated

Nuclear fuel rods.

Modeling

1. Makes nuclear fuel pin steady-state performance predictions and obtains material properties for input into other codes.
2. The solution method consists of an iterative convergence for each axial fuel node and an iterative convergence on the total fuel pin internal gas inventory for each time-power step.

Limitations

1. Allows a maximum of 15 axial nodes and 20 time-steps. The first time-step must be zero. The second time-step cannot be at zero power.

Computer Operations

1. Operable on the CDC 6600 computer using Scope 3.4.2. Written in FORTRAN IV.
2. Requires 60,000 (octal) words of core memory.
3. Typical run time is about 1 to 3 minutes on the CDC 6600.
4. Consists of 20 subroutines, 6 functions, a comdeck, and the main GAPCON program.

Reference

C. E. Beyer, C. R. Hann, D. D. Lanning, F. E. Panisko and L. J. Parchen, GAPCON-THERMAL-2: A Computer Program for Calculating the Thermal Behavior of Oxide Fuel Rods, BNWL-1897 and BNWL-1898, November 1975.

[redacted]

GLUB 1

Purpose

GLUB 1 solves the water-logging transient caused by the existence of fuel element cladding defects during an increase in power.

Status

A.	Development	Developed by the Westinghouse Electric Corporation at the Bettis Atomic Power Laboratory, West Mifflin, Pennsylvania.
B.	Availability	Released to the National Energy Software Center (NESC) in 1969. Restricted to use by U.S. citizens for either official U.S. Government or commerical nonmilitary applications at a U.S. installation. Cost $64.00.
C.	Utilization	Used to evaluate reactor start-up transients due to cladding defects and to indicate maximum power before cladding rupture occurs.

Problems Solved

Calculates internal pin pressure, defect flow, cladding stress versus time, and steady-state power that causes cladding failure.

Systems and Subsystems Simulated

Nuclear fuel rod and associated coolant channel.

Modeling

1. Calculations are performed for two cases:
    a. Power as an arbitrary function of time, and
    b. Power increases assumed to be a series of ramps with a steeper and steeper slope and ending at a new steady-state power.
2. The transient is computed as a sequence of quasi-steady states with fuel and clad temperatures, stress and strains being iteratively computed at 11 different values of volume averaged enthalpy. Linear interpolation then gives these values at any other volume averaged enthalpy. Strains and enthalpy are used to calculate the fuel element core, plenum and annulus volumes, and enthalpies versus time. Stress and strain calculations are done elastically, allowing for interference between fuel and clad.

3. The fuel element model is split into two parts, the fuel section and the plenum. The split occurs axially at the top of the fuel. The fuel section contains control volumes for fuel, core (if one exists), annulus, and clad. The plenum section has control volumes for plenum and clad.

4. All the geometry and metal properties are input quantities.

Limitations

1. The code allows a maximum of:
    a. 10 metal property versus temperature pairs for each metal input.
    b. 30 power or defect enthalpy versus time pairs.
    c. 10 defect diameters to determine allowable power as a function of defect diameter.

2. Assumes heat convection due to water flowing between sections and out defect as negligible.

Computer Operations

1. Operable on the CDC 6600 using Scope. Written in FORTRAN IV.

2. Requires 60,000 (octal) words of storage on the CDC 6600.

3. Typical run time is 5 minutes.

References

F. T. Dunckhorst, L. L. Lynn, W. A. Coffman, and J. E. Meyer, GLUB-1 - A FORTRAN IV, Digital Computer Program for Water-Logged Fuel Element Analysis, WAPD-TM-569, November 1966.

C. J. Pfeifer, CDC 6600 FORTRAN Programming - Bettis Environmental Report, WAPD-TM-6688, January 1967.

[redacted]

GRASS

Purpose

GRASS performs analysis of steady-state and transient gas release and swelling of oxide fuels.

Status

A.	Development	Developed and under improvement at Argonne National Laboratory, Material Science Division, Argonne, Illinois.
B.	Availability	Operational.
C.	Utilization	Developed to serve as a subroutine to the LIFE code but may operate independently. Has been used to predict steady-state and transient out-of-pile irradiation experiments on LWR fuels.

Problems Solved

Predicts the fission gas release (i.e., movement of fission gases to and from grain boundaries and grain edges and/or to the fuel central region) and fuel swelling (i.e., fuel volumetric expansion due to fission gas accumulation and subsequent fuel displacement) as a function of time and operating conditions.

Systems and Subsystems Simulated

Nuclear oxide fuel.

Modeling

Predictions are based on the following models of microscopic fission gas behavior:

1. Bubble migration (motion by all competing mechanisms),
2. Bubble coalescence and growth,
3. Nonequilibrium effects,
4. Bubble irradiation-induced re-solution,
5. Bubble interaction with structural defects (i.e., grain boundaries and dislocations), and

6. Channel formation on grain boundaries and the evolution of interconnected porosity along the grain edges.

These models are used to compute the evolution of bubble-size distribution for all types of bubbles in the fuel; this, in turn, is used to compute total fuel swelling and gas release. All equations are one-dimensional.

Limitations

1. Allows a maximum of 15 fuel regions.

2. Requires input of fuel temperature, temperature gradient, fission rate, fuel density, grain size, and hydrostatic stress state.

Computer Operations

Operable on the IBM 370.

Interface/Overlap

The SAS, LIFE, and FRAS codes perform similar calculations.

References

R. B. Peoppel, "An Advanced Gas Release and Swelling Subroutine," Proc. Conf. on Fast Reactor Fuel Element Tech., American Nuclear Society (1971), pp. 311-326.

J. Rest, M. G. Seitz, S. M. Gehl, L. R. Kelman, "Development and Experimental Verificaton of SST-GRASS: A Steady State and Transient Fuel Response and Fission-Product Release Code," to appear in Proc. OECD-CSNI Mtg. on the Behavior of Water Reactor Fuel Elements Under Accident Conditions, Norway (September 1976).

J. Rest, "Light-Water-Reactor Safety Research Program: Quarterly Progress Reports."

R. B. Peoppel and Pati, Trans. AND, *15* (June 1972).

J. W. Dias and R. B. Peoppel, *Transient Swelling Studies with the GRASS Code*, ANL-7992 (March 1973).

E. E. Gruber, "Calculated Size Distributions for Gas Bubble Migration and Coalescence in Solids," *J. of Applied Phys.*, 38, 243 (January 1967).

Harrison, *JNM*, *31* (1969).

V. Z. Jankus and R. W. Weeks, *LIFE1, A FORTRAN IV Computer Code for the Prediction of Fast Reactor Fuel Element Behavior*, ANL-7736 (November 1970).

E. E. Gruber, *Transient Gas Release from Oxide Fuels: Parametric Representation of FRAS Results*, ANL/RAS 75-7 (March 1975).

M. G. Stevenson, W. R. Bohl, F. E. Dunn, T. J. Heames, G. Hoppner, and L. L. Smith, "Current Status and Experimental Basis of the SAS LMFBR Accident Analysis Code System," *Fast Reactor Safety*, USAEC-Conf. 740701, Beverly Hills, California (April 1974).

[redacted]

HOTPIN

Purpose

HOTPIN represents a one-dimensional, two-phase compressible hydrodynamic model for the motion of a molten fuel/fission gas mixture inside a fuel pin following cladding rupture during a hypothetical transient overpower accident.

Status

A. Development	Developed by Hanford Engineering Development Laboratory (HEDL), Richland, Washington, for inclusion in the MELT code.
B. Availability	Operational at HEDL on the CDC 6600. The original version has been incorporated into MELT-III, and an improved version is being incorporated into MELT-IIIA.
C. Utilization	Used by HEDL to do sensitivity analyses on molten fuel dynamics for fast flux test facilities (FFTFs). Primarily, a fast reactor code but can be modified for LWRs.

Problems Solved

1. Calculates nonuniform pressure, densities, and velocities as a function of time and length along the fuel pin.

2. Calculates rupture size and the amount and composition of the molten fuel/fission gas mixture that escapes into the coolant channel.

Systems and Subsystems Simulated

Nuclear fuel rod melting dynamics and two-phase flow.

Modeling

1. Uses the Hartrees method of characteristics, which has been modified to allow anisotropy in one-dimensional, two-phase fluid flow.

2. The mass, momentum, and energy conservation equations are reduced to characteristic form and solved simultaneously on a constant grid by using an iterative scheme.

3. The models include:

   a. Homogeneous, one-dimensional flow with a bubble drift model for relative motion,
   b. Two-phase, two-component fluid with no mass transfer, and
   c. Consideration of only bubbly flow regimes.

Limitations

1. Requires internal pin conditions at time of clad rupture (e.g., dimensions of the molten region, initital void fraction, and heating rate). The location and size of the rupture and the change in the dimensions of the molten region as a function of time are also required as input.

2. Does not account for condensation and evaporation effects.

Computer Operations

Information not readily available.

Interface/Overlap

HOTPIN has been incorporated into the MELT-III code.

References

H. J. Willenberg and A. Padilla, Jr., "Analysis of Transient Compressible Two-Phase Flow With Heat and Mass Sources Using the Method of Characteristics," Proceedings of the Conference on Computational Methods in Nuclear Engineering, CONF-750413, Volume 1, Charleston, South Carolina, April 15-17, 1975.

H. J. Willenberg, Improvements in HOTPIN Internal Fuel Motion Code, Fast Reactor Safety Technology Progress Report--April-May-June, 1976, HEDL-TIME 76-56 (Richland, Washington: Hanford Engineering Development Laboratory, March 1977).

A. E. Walter et al., MELT-III, A Neutronics, Thermal Hydraulics Computer Program for Fast Reactor Safety Analysis, HEDL-TIME 74-47 (Richland, Washington: Hanford Engineering Development Laboratory, October 1974).

D. R. Hartrees, Numerical Analysis, 2nd ed. (New York: Oxford University Press, 1958).

LION4

Purpose

LION4 calculates one-, two-, and three-dimensional transient and steady-state temperature distributions in reactors and reactor plant components.

Status

A. Development	Developed by the General Electric Company at the Knolls Atomic Power Laboratory, Schenectady, New York.
B. Availability	Released to the National Energy Software Center (NESC) in 1967. Restricted to use by U.S. citizens for either official U.S. Government or commercial nonmilitary applications at a U.S. installation. Cost $68.00 (CDC 6600 version).
C. Utilization	Primarily used for thermal-structural analysis of reactor components.

Problems Solved

Determines the one-, two-, and three-dimensional temperature distributions in reactor and reactor plant components.

Systems and Subsystems Simulated

Reactor and reactor plant components.

Modeling

1. Modeling is based on the concept of nodal points connected by the one-dimensional thermal conduction equations in as many as six directions, simultaneously.
2. An explicit or first forward difference method is used to obtain the solution of the equations for the three-dimensional field.
3. The temperature change for a surface node is calculated by applying Ohm's Law Analogy for heat flow to a node with no capacitance.
4. Heat transfer includes forced and free convection in both reduced and fully automated forms, coolant flow effects, limited thermal

radiation capability, stationary or stagnant fluid gaps, fuel temerature dependent and temperature level heat transfer, alternative heat transfer mode comparison and selection facility combined with a heat flux direction sensor and any form of time-dependent boundary temperatures. Subcooled nucleate boiling may be considered but is limited to water.

5. Handles space- and time-dependent heat generation and can also provide temperature-dependent material properties with limited nonisotropic properties.

6. Available user-oriented capabilities include temperature means with various weightings and a complete heat flow surveillance system.

Limitations

The more recent CDC 6600 edition requires

11*D1 + 26*D2 + 20*D3 + 3*D4 + 17*D5 + 3*D6 + S1 + S2 + S3 + S4
less than or equal to 245000,

where

D1 = number of external connections,
D2 = number of internal nodes,
D3 = number of surface nodes,
D4 = number of boundary nodes,
D5 = number of coolant nodes,
D6 = number of interface nodes,
S1 = storage for temperature means and differences,
S2 = storage for the heat flow subroutine,
S3 = storage for convective heat transfer calculation, and
S4 = storage for variable libraries.

Using an increased nodal capability feature with an admittance matrix in ECS, this algorithm becomes:

11*D1 + 10*D2 + 4*D3 + 3*D4 + 17*D5 + 3*D6 + S1 + S2 + S3 + S4

The January 1970 edition requires

11*D1 + 25*D2 + 20*D3 + 62*D4 + 15*D5 less than or equal to 21000.

for

15 materials,
7*D2 internal connections,
12 coolant channels (each with 4 sides maximum),
77 printouts,
7 internal connections per internal node,
6 surface connections per surface node,
6 types of temperature means and differences sets,

10 graphs (CDC 6600, 7600 only),
12 curves per graph (CDC 6600, 7600 only),
17 types of quantities (CDC 6600, 7600 only),
4 subcooled nucleate boiling regions, and
5 types of heat flow rate sets.

Computer Operations

1. Operable on the CDC 6600 and 7600 using Scope, on the UNIVAC 1108 A,B using EXEC 8, and on the IBM 360 and 370 using OS/360 and 370, respectively. Written in FORTRAN IV and ASCENT (CDC 6600,7600) and FORTRAN IV (UNIVAC 1108 A,B and IBM 360,370).

2. Uses dynamic allocation. The IBM 360,370 version requires 350,000 for program execution and 1 additional input/output unit besides the standard unit. The CDC 6600 version plotter routine Apple is used to produce the input required by the associated Calcomp plotter for graphical output.

3. Run time on the CDC 6600 is strongly dependent upon the number of nodes, the relative and absolute size, and the time range considered. The sample problem executed in a few seconds on the CDC 6600, in about 5 seconds on the UNIVAC 1108 A, and in about 2 minutes on the IBM 360.

Interface/Overlap

Support programs include

1. PLI, a series of preprocessors of selected geometrics,

2. GRAPL, a gradient plotter,

3. HCAL2, which evaluates heat transfer coefficient equations for selected conditions,

4. CONLI, which renumbers input data and inserts spare node numbers of three different kinds to permit consolidation of geometry input for entry into a large program, and

5. KAPLOT, for graphical output on the CDC 6600.

LION supercedes earlier IBM 704 and Philco 2000 programs called TIGER.

References

J. R. Schmid, G. L. Lechliter, and W. W. Fischer, Lion Temperature Distributions for Arbitrary Shapes and Complicated Boundary Conditions, KAPL-M-6532, July 27, 1966.

W. W. Fischer, Lion Tales - A Users Manual for the Lion Thermal-Structural Evaluation Code, KAPL-M-6533, July 31, 1967.

E. J. Binney, Addendum to Lion, KAPL-M-6532, revised January 1969.

E. J. Binney, Additions to Lion Code, KAPL Note, September 30, 1969.

E. J. Binney, Additions to Lion Code Effective with September 1970 Version, KAPL Note, September 22, 1970.

W. W. Fischer and M. N. Goldsmith, Graph, A Gradient Plotter for Lion, KAPL-M-7164(L), October 13, 1970.

E. J. Binney, Additions to Lion Code Effective with the December 1970 Version, KAPL Note, December 18, 1970.

E. J. Binney, Additions to the Lion Code Effective February 26, 1971, KAPL Note, February 26, 1971.

E. J. Binney, Modification to the Lion Code Effective March 31, 1971, KAPL Note, April 8, 1971.

E. J. Binney, Heat Balance Capability in Lion, KAPL Note, April 30, 1971.

E. J. Binney, Additions to the Lion Code Effective February 29, 1972, KAPL Note, February 29, 1972.

E. J. Binney, Lion Newsletter No. 1, KAPL Note, June 5, 1972.

E. J. Binney, Lion4 Newsletter No. 2, KAPL Note, October 5, 1972.

M. N. Goldsmith and W. W. Fischer, The PLI--Series of Preprocessors for the Lion Code, KAPL-M-7289(L), July 6, 1972.

"JPARAM Systems Programming Informal Program Description," Excerpt for Users Guide to CDC Systems at KAPL, July 27, 1972.

W. W. Fischer, Convective Heat Transfer Coefficient Calculational System for Lion, KAPL-M-7301(L), December 18, 1972.

KAPL Internal Calcomp Subroutines, KAPL Note, March 31, 1971.

A. Peronilla, LION IV, IBM System/360 System/370 Version, Foster Wheeler Note, January 29, 1974.

G. Krevolin, Lion IV, UNIVAC 11088 Version and Addendum to LION4, June 22, 1973.

[redacted]

LOFTRAN

Purpose

LOFTRAN computes the detailed transient behavior of a one-loop PWR system during anticipated transients without scram (ATWS), operational transients, and steam generator tube ruptures.

Status

A.	Development	Developed by the Nuclear Energy Systems Division of the Westinghouse Electric Corporation, Pittsburgh, Pennsylvania.
B.	Availability	Proprietary.
C.	Utilization	Used to analyze various accident responses and control studies. Used to comply to NRC request on ATWS behavior of Westinghouse PWR plants.

Problems Solved

Determines transient fluid thermal-hydraulic conditions throughout the system and transient thermal conditions of the core material (fuel rods and structure) as a result of an accident or control function.

Systems and Subsystems Simulated

One loop PWR reactor system including pump, core with internals, steam generator, and pressurizer.

Modeling

1. The geometrical system is modeled with control volumes and junctions. The fuel rods are modeled by an axial segment for one radial zone.
2. Assumes uniform pressure within a volume, homogeneous flow except in the core regions, no reverse flow through a control volume, and isentropic compression and expansion.
3. The core flow can be either calculated from the momentum equation, a flow coastdown of hyperbolic form, or an arbitrary time function supplied by the user.
4. The reactor core and fuel rod temperatures are calculated, and the heat flux can be calculated or user supplied.

5. The core flow of steam and water are homogenized in the upper plenum and core outlet regions.

6. Piping flow considers homogeneous slugs moving through a volume.

7. The steam generator primary to secondary heat transfer is considered with provision for tube rupture and transients in feedwater flow with pressure effects on load.

8. The pressurizer is considered with the effects of spray flow, heaters, condensation, flashing, and safety and relief valves.

9. The emergency core cooling (ECC) system contains high-pressure accumulators and low-pressure systems with boron concentration in the coolant.

10. Reactor kinetics uses a point kinetics equation with six delay neutron groups and computes nuclear power, including reactivity effects from control rod motion, rod trip, Doppler, and moderator density. Reactivity coefficients are user supplied.

11. A control system module is used that includes control rod control, steam dump, and feedwater and pressurizer pressure controls.

12. The following inputs are required:

    a. Reactor coolant fluid volumes,
    b. Flow fraction, auxiliary system volumes,
    c. Core average temperature and nominal full-power fluid conditions,
    d. Fuel and clad characteristics,
    e. Secondary system parameters,
    f. Safety valve characteristics,
    g. Reactor coolant pump characteristics,
    h. Reactor coolant loop pressure drops at design full power, and
    i. Flow areas and dimensions.

Limitations

1. The number of control volumes and junctions is limited by the computer capacity.

2. The fuel rod calculations consider one radial node and a maximum of 40 axial nodes in the core.

3. No core cross-flow effects are considered.

4. The fuel rod to coolant heat transfer considers the cladding at the coolant temperature, and, if the fuel enthalpy is greater than the heat transfer coefficient indicates, a constant film temperature is assumed. This limits the calculation of critical heat flux, transition boiling, and rewet.

5. Best suited for operational transients and ATWS. The slug flow assumption does not allow accurate predictions of the LOCA transient.

Computer Operations

Operable on the CDC 6600. Written in FORTRAN IV.

Reference

T. W. T. Burnett, C. J. McIntyre, and J. C. Buker, *LOFTRAN Code Description*, WCAP-7907, October 1972.

[redacted]

MARVEL

Purpose

MARVEL was developed to calculate the detailed multiloop transient behavior of PWRs. It calculates the behavior of operational transients, anticipated transients, and some small-break (or leak) transients.

Status

A. Development	Developed by Westinghouse Electric Corporation, Nuclear Energy Systems Division, Pittsburgh, Pennsylvania.
B. Availability	Proprietary.
C. Utilization	Can be used as a tool for various types of accident analysis and control studies.

Problems Solved

Calculates the fluid thermal and hydraulic behavior within the system during the transient and also determines the behavior of the reactor core nuclear and thermal kinetics.

Systems and Subsystems Simulated

The reactor system conforms to systems designed for current Westinghouse PWRs with and without preheat-type steam generators. MARVEL simulates two reactor coolant loops, including steam generators, pressurizer, reactor vessel, and associated systems. Also simulates reactor kinetics, reactor control and protection systems, and safeguard systems.

Modeling

1. The geometry of the system is described by nodes or flow sections. The reactor coolant loops may have loop isolation valves, safety injection system connections in the hot legs and cold legs, and charging and letdown connections.

2. In each node, the mass and energy balance equations are solved by integration over time using the finite difference method.

3. In the reactor vessel, mixing in the upper and lower plenums is considered along with cross flow in the core region. The core nuclear and thermal kinetic models assume a cosine-shaped axial neutron and heat flux shape, and the coolant properties are calculated by a flux squared weighting. The kinetic model considers an

instantaneous fission rate and contributions from six delayed neutron groups. The rate of change of fission power is calculated by the conventional one-energy space independent kinetics equation.

4. Six different flow models are present for coolant system thermal kinetics equations: transport delay, mixing, steam generator heat transfer, heated core, reactor vessel outlet plenum, and an inactive coolant volume.

5. Models are provided for natural circulation, the rod control system, the steam dump control system, the overtemperature and overpower protection system, the safety injection system, and the chemical and volume control system.

6. The program can simulate the following perturbations or transients:

   a. Steam line breaks,
   b. Feedline breaks,
   c. Small breaks in the reactor coolant system,
   d. Reactivity perturbations,
   e. Core power fluctuation,
   f. Reactor coolant loop isolation valve movement,
   g. Reactor coolant loop flow fluctuation,
   h. Steam and feedwater flow perturbations,
   i. Feedwater enthalpy perturbations,
   j. Secondary loop isolation valve movement,
   k. Pressurized auxiliary spray perturbations,
   l. Reactor trip,
   m. Safety injection system operation,
   n. Chemical and volume control operation, and
   o. Steam dump system operation and others.

Limitations

1. The program is set up for Westinghouse PWR plants.

2. The geometric model has a maximum of 43 nodes in the specific arrangement given below:

   a. Six hot leg nodes (3 in each loop),
   b. Six cold leg nodes (3 in each loop),
   c. Two steam generator inlet leader nodes (1 in each loop),
   d. Four steam generator tube nodes (2 in each steam generator),
   e. Two steam generator outlet nodes (1 in each loop),
   f. Four vessel inlet plenum nodes,
   g. Eight reactor core nodes arranged in two parallel flow channels,
   h. One core bypass node,
   i. Two reactor upper plenum nodes below outlet nozzles,
   j. One reactor vessel outlet node,
   k. One reactor vessel inactive volume, and
   l. Two pressurizer nodes (one for the vapor and one for the liquid).

[redacted]

3. The program is designed to run within temperature limits of within 50 to 1,100°F and pressure limits of within 10 to 7,000 psia.

Computer Operations

Operable on the CDC 7600 computer. Written in FORTRAN IV.

Reference

R. C. Krise and S. Miranda, MARVEL - A Digital Computer Code for Transient Analysis of a Multiloop PWR System, WCAP-8844, November 1977.

[redacted]

MICE

Purpose

MICE performs analysis of multifield fluid flow and structural responses in a containment system during a hypothetical core disruptive accident.

Status

A.	Development	Under development at Argonne National Laboratory (ANL), Argonne, Illinois.
B.	Availability	Still in its early stage of development. The present version, which solves two material fields fluid flow with material interpenetration and structural responses, is operational and undergoing testing at ANL on the IBM 370.
C.	Utilization	Has been applied to several sample problems with two material fields fluid flow in a containment system. The reactor wall and core barrel are treated as flexible structures that undergo elastic-plastic deformations.

Problems Solved

Solves multifield fluid flow and structural responses in a containment system during core disassembly. After completion of the code, phenomena such as phase transition due to sodium boiling and heat transfer between material fields will be modeled.

Modeling

1. The MICE code uses 2-D axisymmetrical cylindrical coordinates. The mesh is evenly divided in radial and axial directions.

2. Fluids in each region in a containment system are modeled by several material fields with different volume fractions.

3. Structures in the containment system are represented by either rigid obstacles or flexible shells.

4. The numerical technique used for the solution of the multifield fluid flow consists of two principal steps. In the first step, the material fields in each cell are treated independently; in the second step, a pressure equilibration is performed among all fields in each cell.

Limitations

1. Heat transfer between material fields is not treated.
2. Phase transition is not calculated.
3. Mesh size is limited to 20 and 50 in radial and axial directions, respectively.

Computer Operations

1. Operational on the IBM 370. Since MICE is written in FORTRAN, it should be operational on the CDC or other computer machines after removal of the double-precision statement in the main program and in each subroutine.
2. Core memory requirement for a mesh size of 20 by 50 and two material fields is about 275,000.
3. Run time for each problem depends strongly upon the volume fraction in each cell and the time-step increments. At present, no definite criterion has been established.
4. Verification of the wave propagation in a straight pipe for a single material field has been reported in ANL-RDP-51, 1976, p. 7.56.

Future Efforts

1. Incorporation of a heat transfer model.
2. Calculation of the phase changes.
3. Incorporation of more types of structural elements.

References

No documentation specific to the MICE code is currently available. The following references discuss the numerical techniques employed in the code:

F. H. Harlow and A. A. Amsden, "Flow of Interpenetratine Material Phases," J. Comp. Phys, 18, 440 (1975).

F. H. Harlow and A. A. Amsden, "Numerical Calculation of Multiphase Fluid Flow," J. Comp. Phys., 17, 19 (1975).

T. Belytschko and B. J. Hseih, "Non-Linear Transient Finite Element Analysis with Convected Coordinates," International Journal for Numerical Method in Engineering, 7, 255 (1973).

J. R. Travis, F. H. Harlow, and A. A. Amsden, *Numerical Calculation of Two-Phase Flow*, LA-5942-MS (Los Alamos: Los Alamos Scientific Laboratory, 1975).

A. V. Kalinin, *Derivation of Fluid-Mechanics Equations for a Two-Phase Medium with Phase Changes*, Heat Transfer-Soviet Research *2*, 83 (1970).

R. I. Nigmatulin, "Equation of Hydrodynamics and Compression Shock in Two-Velocity and Two-Temperature Continuum with Phase Transformations," *Fluid Dynamics*, *2*, 20 (1967).

F. H. Harlow and A. A. Amsden, "A Numerical Fluid Dynamics Calculation Method for All Flow Speeds," *J. Comp. Phys.*, *8*, 197 (1971).

[redacted]

MO219 (FLOT1)

Purpose

MO219 (FLOT1) predicts the steady-state flow and flow transient due to the subsequent loss of power to some or all of the primary coolant pumps in a PWR power plant.

Status

A.	Development	Developed by the Westinghouse Electric Corporation at the Bettis Atomic Power Laboratory, West Mifflin, Pennsylvania.
B.	Availability	Released to the National Energy Software Center (NESC) in 1968. Restricted to use by U.S. citizens for either official U.S. Government or commercial nonmilitary applications at a U.S. installation. Cost $64.00.
C.	Utilization	Used for core performance analysis and to check valve pressure surge calculations.

Problems Solved

Predicts transient flow distribution and both pre- and post-incident steady-state flow distributions in piping and reactor components. Also predicts other thermal hydraulic parameters.

Systems and Subsystems Simulated

Nuclear plant primary coolant system and primary coolant pumps.

Modeling

1. The simultaneous differential equations of flow are solved by a matrix method and, in conjunction with the dynamic equations for the pump, are iterated to user-specified convergence criterion.
2. Empirical relations for the transfer of energy from the pump impeller to the fluid over the entire range of operations have been included in the program, and the results show good agreement with test data for the types of test pumps considered.
3. The transient may be terminated by a maximum transient time or by check valve closures in all loops in which pumping power is lost.

Limitations

1. Limited to four piping loops with a maximum of three pumps per loop in parallel.

2. Empirical relations for impeller-to-fluid energy transfer limit applicability to

    a. Pumps of a design and size similar to those used to obtain the empirical data, and
    b. Pumps for which data are available to construct new empirical relations.

Computer Operations

1. Operable on the CDC 6600 using Scope 2.0. Written in FORTRAN IV.

2. Requires 55,000 octal locations with card batch input and both printed and plotted output.

3. Run time for a three-loop, initial operation to complete loss of flow requires 49 seconds with output consisting of initial steady-state flow and pressure drop distribution and a printout every .05 second for 10 seconds.

References

G. M. Fuls, FLOT-1 Flow Transient Analysis of a Pressurized Water Reactor during Flow Coastdown, WAPD-TM-428, April 1968.

C. J. Pfeifer, CDC6600 Fortran Programming - Bettis Environmental Report, WAPD-TM-668, January 1967.

[redacted]

NATRANS 2

Purpose

NATRANS 2 analyzes short-term, pressure-pulse transients in closed hydraulic systems.

Status

A.	Development	Developed by Argonne National Laboratory, Component Technology Division, Argonne, Illinois.
B.	Availability	Released to the National Energy Software Center (NESC) in 1975. Cost $64.00.
C.	Utilization	Used to analyze the pressure-pulse transient resulting from a sodium-water reaction in a liquid metal fast breeder reactor (LMFBR); however, other working fluids and transient pulses can be analyzed.

Problems Solved

Determines pressure propagation and other hydraulic parameters for the system so that structural analysis can be performed.

Systems and Subsystems Simulated

A two-dimensional axisymmetric domain (reactor vessel, steam generator, or other vessels) connected to a one-dimensional piping network.

Modeling

1. The one-dimensional piping network may consist of series or parallel piping, pipe juctions, diameter changes, junctions of three to six branches, closed ends, surge tanks, far ends, dummy junctions, acoustic impedance, discontinuities, and rupture disks.
2. The user supplies input for the pressure-time function at some point in either the one- or two-dimensional domain. A sodium-water reaction pressure pulse that follows the dynamic model of Zaher and Solmon can also be used.
3. The default working fluid is sodium; however, other fluids can be specified in terms of density, sonic velocity, and viscosity.
4. The code uses a numerical integration technique based on the method of characteristics and formulated for the system of quasi-linear hyperbolic equations describing fluid-hammer phenomena in a closed two-dimensional axisymmetric domain.

Limitations

1. The code provides for a maximum of

    a. 1 two-dimensional cylindrical region,
    b. 30 grid points in the r-direction,
    c. 60 grid points in the z-direction,
    d. 50 one-dimensional pipe branches,
    e. 120 nodes per pipe branch,
    f. 50 junctions, and
    g. 10 surge tanks.

2. Cavitation is assumed to be absent.

3. In the two-dimensional domain, viscosity effects are neglected.

4. A sufficiently large length-to-diameter ratio must be allowed for extended regions both upstream and downstream of the two-dimensional domain since matching of boundary conditions at the interfaces neglects two-dimensional effects.

Computer Operations

1. Operable on the IBM 360 and 370 using OS/360 and 370, respectively. Written in FORTRAN IV.

2. Requires 452,000 bytes of storage and a disk or tape unit for plotting.

3. Run time is dependent upon the complexity of the problem but requires approximately 100 to 200 seconds of CPU time.

Interface/Overlap

The Natransient code is used as a subroutine for the one-dimensional system calculation. Related programs are WHAM (p. A-137) and COMPARE (p. A-23).

Reference

Y. W. Shin and R. A. Valentin, Two-Dimensional Fluid-Hammer Analysis by the Method of Characteristics in a Closed Axisymmetric Cylindrical Domain, ANL-8090, August 1974.

[redacted]

ODYN

Purpose

ODYN calculates the operational transient behavior of a BWR power plant.

Status

A.	Development	Under development by the General Electric Company (GE), San Jose, California.
B.	Availability	Proprietary. Code is still in development and verification stages.
C.	Utilization	Used to calculate BWR power plant transient performance prior to plant operation and for licensing calculations. Has been submitted to NRC for licensing.

Problems Solved

1. Calculates the thermal-hydraulic parameter within a BWR plant core, recirculation, and secondary systems. The code also calculates fluid parameters and some component thermal parameters.

2. Some of the most common safety transients studied are

   a. Loss of turbine generator node,
   b. Failure of feedwater system,
   c. Reactor scan,
   d. Recirculation pump trip,
   e. Pressure relief system response,
   f. Pressure regulator system failure, and
   g. Control system design transients.

Systems and Subsystems Simulated

BWR power plant integrated core, recirculation system, secondary system, and major control systems.

Modeling

1. The neutron flux distribution is calculated assuming time-dependent spatial variation only in the axial direction. One energy group diffusion theory is used. Six delayed neutron groups are assumed. The neutron diffusion parameters are obtained from collapsed three-dimensional steady-state conditions.

2. The core average axial water density and pressure distribution is calculated using a single channel to represent the heated active flow and a single channel to represent the bypass flow. A five-equation model, representing liquid and vapor mass and energy conservation and mixture momentum conservation, is used to describe the thermal-hydraulic behavior. Thermal-hydraulic correlations consistent with accepted GE design procedures are used. Pressure and saturated properties are allowed to vary axially.

3. Heat transfer to the moderator and fuel temperature are calculated assuming an average cylindrical fuel and cladding model for each axial location in the core. Axial and time variations in gap conductance may be input. The conduction parameters are temperature dependent.

4. Decay heat is modeled through a simple exponential decay model.

5. Changes with the flow split between the bypass and active channel flow during the transient are accounted for. Core pressure drop is normalized to an initial input value.

6. Pressure variations in the primary system are described with 10 pressure nodes, including nodes in the upper plenum, dome, and 8 nodes in the main and bypass steam lines.

7. Energy and mass balances are carried out for the internal centrifugal steam separators.

8. Liquid and vapor mass volume balances are used to predict the reactor vessel water level changes. Perfect mixing is assumed between the saturated liquid, carryunder, high-pressure core injection flow, and subcooled feedwater.

9. The independent recirculation loop model can simulate any combination of multiloop systems. The jet pump models have been included with either valve flow control or motor-coupled generator recirculation pump drive units. The jet pump mass and energy balance equations assume perfect mixing between suction and drive flow and in the vessel inlet plenum. Also, all recirculation flow is assumed to be subcooled and incompressible.

10. The turbine pressure controls, feedwater controls, load-following controls, and safety system functions are included in the model and are simulated through digital logic in the transient model. The numerical values for response functions, as well as gains and time constants, must be input for each plant design.

Limitations

1. Designed to handle transients which occur on a time scale of seconds rather than hours or days.

2. The models for prompt critical transients have not been qualified, and such results should be used with caution.

3. The current version of the code does not include correlations for film boiling, which may preclude its use in very severe transients such as anticipated transients without scram (ATWS).

4. The one-dimensional transient model does not have a hot channel capability and hence must be used to drive other transient thermal-hydraulic models to determine hot pin, hot channel characteristics.

Computer Operations

Being developed on the CDC 7600. Written in FORTRAN IV.

Interface/Overlap

Reactivity calculations from the ODYN code can be used as input to the REDY code.

References

"Qualifying of the One Dimensional Core Transient Model for Boiling Water Reactors, Volumes 1 and 2." Licensing Topical Report, NEDO-24154, (General Electric Company: October 1978).

R. B. Linford, Analytical Methods of Plant Transient Evaluations of the General Electric Boiling Water Reactor, NEDO-10802, February 1973.

C. H. Robbins, Performance Tests of Axial Flow Primary Steam Separators, APED-4762, January 1965.

L. A. Carmichael and R. O. Niemi, Transient and Stability Tests at Peach Bottom Atomic Power Station Unit 2 at End of Cycle 2, EPRI NP-564, June 1978.

N. H. Larsen, Core Design and Operating Data for Cycles One and Two of Peach Bottom-2, EPRI-NP563, June 1978.

ORCENT

Purpose

ORCENT performs full load design point heat and mass balance calculations for steam turbine systems and cycles typical of nuclear power plants.

Status

A.	Development	Developed by the Oak Ridge National Laboratory, Oak Ridge, Tennessee.
B.	Availability	Released to the National Energy Software Center (NESC) in 1976. Cost $66.00
C.	Utilization	Used to perform design and efficiency calculations on steam turbines under operation.

Problems Solved

Determines mass flows and heat data for the steam cycle in a turbine characteristic of a nuclear power plant.

Systems and Subsystems Simulated

Steam turbine and generator including stages, inlet steam section, outlet flow section, condenser, and feedwater heaters.

Modeling

1. A lumped parameter model is use for the physical system.
2. Turbine performance calculations are based on the General Electric Company method for large steam turbine-generators operating with saturated and low superheat throttle steam. The output includes all information normally shown on a turbine cycle heat balance diagram.
3. Handles inert throttled steam and both condensation and back pressure turbine exhaust arrangements.
4. The efficiency calculational method is based on detailed consideration of the effect of expansion moisture on stage efficiency.

Limitations

1. A maximum of 5 moisure removal stages in the low-pressure turbine section and 12 feedwater heaters is allowed.

2. One, two, or three turbine sections in series (tandem-component arrangement only) may be specified. Turbine speed should be 1,800 or 3,600 rpm.

3. Throttle steam temperature should be between 0° to 200°F above saturation temperature. Throttle steam pressure should be between 200 and 2,000 psia.

4. The generator rating may be between 100 and 1,500 MVA.

Computer Operations

1. Operable on the IBM 360 and 370 using OS/360 and 370, respectively. Written in FORTRAN IV.

2. Requires 166,000 bytes of storage needed for execution on the IBM 370/195.

3. Run time for a single case is approximately 1 second on the IBM 360/91.

References

H. I. Bowers, ORCENT: A Digital Computer Program for Saturated and Low Superheat Steam Turbine Cycle Analysis, ORNL-TM-2395, January 1969.

F. G. Baily, K. C. Cotton, and R. C. Spencer, Predicting the Performance of Large Steam Turbine-Generators Operating with Saturated and Low Superheat Steam Conditions, GER-2454A, American Power Conference, Chicago, April 25 to 27, 1967.

PAD

Purpose

PAD is used in analyses of destructive and nondestructive reactivity experiments. Nuclear excursions in various fissioning systems as well as dynamic, nonfissioning systems, are described with interactive neutronics, thermodynamics, and hydraulics.

Status

A.	Development	Developed at Los Alamos National Laboratory, Los Alamos, New Mexico.
B.	Availability	Operational at Los Alamos National Laboratory on a CDC 7600 and at the University of Arizona on a CDC 6400. Released to the National Software Information Center (NSIC) in 1977.
C.	Utilization	Used by Los Alamos National Laboratory to analyze destructive and nondestructive experiments involving reactivity transients, including prompt-critical reactivity insertions.

Problems Solved

1. Calculations of temperature rise, recriticality, and total fissions of nondestructive reactivity insertions are performed.
2. Calculations of explosion energy release and thresholds and recriticality for destructive reactivity insertions are performed.
3. Calculations of water hammer, conversion of internal energy to kinetic energy, and sensitivities for several varied reactor parameters to provide bounding estimates of explosive energy release can be performed.

Systems and Subsystems Simulated

Nuclear core, fissioning systems, and dynamic nonfissioning systems.

Modeling

1. A one-dimensional model with options of spherical, planar, or cylindrical systems is used.
2. Neutron kinetics with up to six delayed neutron groups is considered. Special power distribution, material motion reactivity feedback, and temperature feedback are included with spatial reactivity weighting options.

3. In all geometries, the coordinated system is Lagrangian.

4. Two materials for each mass point are allowed with optional heat transfer rates between these materials.

5. The equations of state consist of a set of functions and coded decisions which, with specific constraints, describe a material in the condensed and vapor states, and provide an orderly transition between these states.

Limitations

1. Limited to one-dimension calculations.

2. Only two materials per mass point are considered.

3. No heat transfer between mass points is allowed.

Computer Operations

Operable on the CDC 6600 and CDC 7600. Written in FORTRAN IV.

References

W. R. Stratton et al., "Reactor Power Excursion Studies," given at the International Conference on Engineering of Fast Reactors for Safe and Reliable Operation, Karlsruhe, October 9-13, 1975. [Mode description.]

C. G. Chezem and W. R. Stratton, RAC, A Computer Program for Reactor Accident Calculations, LAMS-2920 (Los Alamos: Los Alamos Scientific Laboratory, January 1963).

D. Okrent, R. Lazarus, et al., AX-1, A Computing Program for Coupled Neutronics-Hydrodynamics Calculations on the IBM 704, ANL-5977 (Illinois: Argonne National Laboratory, May 1959).

PARET

Purpose

PARET predicts the course and consequences of nondestructive reactivity accidents in small reactor cores.

Status

A. Development	Developed by Aerojet Nuclear Corporation at the Idaho National Engineering Laboratory, Idaho Falls, Idaho.
B. Availability	Released to the National Energy Software Center (NESC) in 1972. Cost $88.00
C. Utilization	Used to predict thermal-hydraulic conditions in small cores as a result of reactivity initiated accidents.

Problems Solved

Calculates the pressure drop across the core, the core thermal response, and the point-wise fluid enthalpies, pressures, and mass flow rates along each channel.

Systems and Subsystems Simulated

Water-cooled small reactor cores, fuel elements, and associated coolant channels.

Modeling

1. The core is modeled with several regions, each of which has different power generation, coolant mass flow rates, hydraulic parameters, and other associated conditions. Each region is represented by a single fuel pin or plate plus its associated flow channel. Each region is also axially sectioned.
2. A transient problem is forced through specification of externally inserted reactivity versus time or average core power versus time.
3. Includes heat transfer, continuous reactivity feedback, and hydrodynamic calculations allowing for coolant flow reversal.
4. Heat transfer in each fuel element is computed on the basis of a one-dimensional conduction solution. Local heat generation is determined from calculated or specified core average power level and spatial weighting factors. Empirical convective heat transfer

correlations are used in estimating the rate of heat transfer at the clad-coolant interface.

5. The hydrodynamics calculations are based on a one-dimensional modified momentum integral model in which equations representing conservation laws are solved in each coolant channel at each time-step. The coolant is allowed to range from subcooled liquid, through the two-phase region, up to and including superheated steam.

Limitations

1. A maximum of four fuel element regions with associated coolant channels and 20 axial sections in each fuel element region is allowed.

2. Employs steady-state heat transfer correlations throughout, which possibly may be unrealistic in certain transient situations. Limited in predicting a thermal crisis by the fact that it uses a steady-state departure from nucleate boiling (DNB) correlation since no transient correlation with demonstrated reliability was developed.

3. Accurate description of hydraulic instabilities depends upon accurate calculation of transient pressure and flow fluctuations. The code is limited in this respect because it employs an incompressible model and a simplified void volume generation equation. The magnitude of local pressures predicted to accompany coolant expulsion is strongly affected by the number of axial segments chosen in the computation. Therefore, the hydrodynamic output should be interpreted as a qualitative indication of a possible crisis rather than a reliable quantitative prediction.

4. Uncertainties lie in the realm of boiling. These uncertainties, coupled with the fact that application of the code to boiling situations has not been fully verified, make it imperative to consider the boiling model in the development stage.

5. Not applicable to either destructive excursions or situations in which there is a space-time effect in neutron flux.

Computer Operations

1. Operable on the IBM 360 using OS/360. Written in FORTRAN IV and BAL.

2. Requires 240,000 bytes of memory and one magnetic tape.

3. Run time to calculate 4 channels for 64 time-steps is about 22 seconds.

Interface/Overlap

PARET is an extension of the CHIC-KIN code (p. B-29).

References

C. F. Obenchain, PARET, A Program for the Analysis of Reactor Transients, IDO-17282, January 1969.

J. A. Redfield, CHIC-KIN, A FORTRAN Program for Intermediate and Fast Transients in Water-Moderated Reactors, WAPD-TM-479, January 1965.

"Change to PARET Program," Aerojet Note, 1972.

"Excerpts" from the NRTS Environmental Subroutine Manual, December 1972.

C. J. Smith and W. A. Cody, ANL Data Routing, ANL Subroutine Description Q0545, February 1968.

[redacted]

PARK1

Purpose

PARK1 performs an analysis of interrelated thermal, hydraulic, and nuclear characteristics in a PWR plant under various steady-state and transient conditions.

Status

A.	Development	Developed by Westinghouse Electric Corporation at the Bettis Atomic Power Laboratory, West Mifflin, Pennsylvania.
B.	Availability	Released to the National Energy Software Center (NESC) in 1977. Restricted to use by U.S. citizens for either official U.S. Government or commercial nonmilitary applications at a U.S. installation. Cost $112.00.
C.	Utilization	Has been used to simulate the overall plant dynamic response for PWR plants.

Problems Solved

1. Calculates total system response to plant maneuvers, loss of load, load changes, complete or sequential loss of primary coolant flow, control rod motion, cold water accidents, and start-up or power range rod withdrawal accidents.
2. Can also simulate protection system performance and anticipated transients with or without scram (ATWS).

Systems and Subsystems Simulated

Entire power plant system and components.

Modeling

1. Steady-state and transient calculations are considered.
2. Permits one and two pass cores with multiple channel regions per pass.
3. Space-time reactor kinetics with decay heat effects is included.
4. One-dimensional mass, energy, and momentum equations and separable (space-time) neutronic equations are included.

[redacted]

5. The solution is calculated by trapezoidal integration of the finite difference equations (central differencing) performed in sequence.

Limitations

1. Considers a maximum of 15 channel regions with up to 21 axial mesh points in each core region, 50 in each coolant pipe, and 31 in each steam generator tube bundle.

2. Allows a maximum of eight leakage paths and four interpass ducts for the core region.

3. Allows a maximum of four primary coolant loops and eight steam generator tube bundles.

4. Reactor kinetics consider 6 delayed neutron groups and up to 11 decay heat groups.

Computer Operations

1. Operational on the CDC 7600 using Scope 1.1 and on the CDC 6600 using Scope 3.1.5. Written in FORTRAN IV.

2. Requires 140,000 (octal) words of central memory and 300,000 words of extended core for the CDC 6600 and 150,000 (octal) words of central memory and 300,000 words of extended core for the CDC 7600.

3. Typical CDC 6600 run time is about 20 seconds for steady-state and 3 to 10 seconds for transient. This amounts to about 0.2 to 0.5 second to execute one time-step.

4. Multisegment loading is required for a 140,000-word field length program.

Interface/Overlap

TOPS (p. B-103).

References

W. G. Clark, J. V. Reihing, Jr., and T. A. Porshing, PARK1, A FORTRAN-IV Computer Program for Nuclear Reactor Power Plants Analysis (LWBR Development Program), WAPD-TM-447(L), revised September 1975.

W. G. Clark, Transient Analysis of Steam Generators in Nuclear Power Plants Using Digital Computer Techniques, M.S. Thesis, University of Pittsburgh, 1965.

Contacts

W. G. Clark and J. V. Reihing, Jr.
Bettis Atomic Power Laboratory
Westinghouse Electric Corporation
Box 79
West Mifflin, Pennsylvania 15122

[redacted]

PELEN

Purpose

PELEN solves a two-dimensional, axisymmetric model of fuel pellets and clad elements throughout a time-history.

Status

A. Development	Developed by Westinghouse Electric Company at the Bettis Atomic Power Laboratory, West Mifflin, Pennsylvania.
B. Availability	Released to the National Energy Software Center (NESC) in 1973. Restricted to use by U.S. citizens for either official U.S. Government or commercial nonmilitary applications at a U.S. installation. Cost $64.00 (IBM), $61.00 (UNIVAC).
C. Utilization	Has been used to analyze fuel pellet and cladding temperature and mechanical response to input conditions.

Problems Solved

Calculates the temperature, stress, and deformation of both the fuel pellet and cladding of a nuclear fuel rod.

Systems and Subsystems Simulated

Nuclear fuel pellet and cladding.

Modeling

1. A heat conduction analysis of both fuel and clad is performed to determine the current temperature field, which is dependent upon the power rate, material properties, and geometry.
2. An incremental stress analysis of both fuel and clad is performed to determine current stress and deformation fields.
3. An analysis of the contact between the fuel pellet and cladding is performed to determine corrected current stress and deformation fields.
4. The geometry is subdivided into triangular cross-section, finite element rings. The field equations are derived by variational principles for heat conduction and stress and solved by matrix inversion techniques.

5. The geometrical configuration is internally updated to include dimensional changes from the reference configuration.

Limitations

1. Considers a maximum of 25 history intervals, of which each can have an arbitrary number of subdivisions.

2. Allows a maximum of 200 finite elements.

Computer Operations

1. Operable on the CDC 6600 and UNIVAC. Written in FORTRAN IV.

2. Requires 140,000 (octal) words of core storage, a printer, and extended core storage units.

3. Run time is about 2.5 seconds of computation per time-step for an 18-finite-element analysis.

4. Requires the INPF (a Bettis input data package).

References

R. B. Stout, E. Duncombe, and J. K. Fischer, PELEN - A Computer Program to Analyze Axial and Radial Variations of Temperature and Deformation in Zircaloy-Clad Oxide Fuel Rods, WAPD-TM-1001(1), December 1971.

C. J. Pfeifer, CDC 6600 FORTRAN Programming - Bettis Environmental Report, WAPD-TM-668, January 1967.

PUMP

Purpose

PUMP determines the flow and pressure distribution in a closed four-pump, two-loop PWR to starting and tripping all combinations of reactor coolant pumps.

Status

A.	Development	Developed by Babcock and Wilcox, Nuclear Power Generation Division, Lynchburg, Virginia.
B.	Availability	Proprietary.
C.	Utilization	Used for design and analysis of components. Has been used to analyze operational transients. Results have been compared to data from the Oconee 1 Nuclear Generating Station.

Problems Solved

Predicts the steady-state and transient response of the flow, pressure, pump speed, and torque relationships.

Systems and Subsystems Simulated

A four-pump, two-loop PWR and reactor coolant pumps.

Modeling

1. The plant is modeled with nodes and flow paths: one node for each pump inlet and outlet, one node for each steam generator junction of hot and cold legs, and two nodes for the reactor inlet and outlet, for a total of 12 nodes.
2. The pump characteristics are derived from pump affinity laws and from vender-supplied pump zone maps (homogeneous pump curves).
3. Any combination of pump coastdown or failure can be simulated. Transient types such as a locked-rotor accident, pump start-up, and transfer function analysis can also be simulated.

Limitations

1. Only the four-pump, two-loop plant can be simulated.

2. The initial conditions are 100% flow with all four pumps operating.

3. The system is assumed to be isothermal throughout all transients (i.e., the fluid density is not sensitive to changes in heat flow caused by changes in flow through the reactor vessel or steam generators). Two-phase flow and two-phase pump operation are not considered.

4. The flow paths through the core are assumed not to change as a function of pump combination and flow dynamics during a transient.

Computer Operations

PUMP is a hybrid (analog-digital) computer code. The digital computer generates the characteristics of the pump and the appropriate friction factors, and the analog computer simulates the remainder of the reactor coolant system.

Interface/Overlap

Used in conjunction with the CADD (p. B-19) and RADAR (p. B-95) codes to calculate operational transients.

Reference

M. R. Grandia, PUMP - Analog-Hybrid Reactor Coolant Hydaulic Transient Model, BAW-10073, July 1973.

RADAR

Purpose

RADAR performs a fuel rod thermal analysis of slow reactor transients such as loss of a primary pump (operational transients).

Status

A.	Development	Developed by Babcock and Wilcox (B&W), Nuclear Power Generation Division, Lynchburg, Virginia.
B.	Availability	Proprietary.
C.	Utilization	Used for design and analysis of reactor coolant conditions during operational transients. Has been used to study anticipated transients without scram (ATWS) behavior of B&W plants in response to NRC licensing and safety criteria. Written primarily for reactor flow coastdown followed by a scram.

Problems Solved

Computes, as a function of time, fuel pin temperatures, cladding surface temperatures, critical heat flux ratios, pressure drops, and coolant thermodynamic conditions in associated fuel pin coolant channels.

Systems and Subsystems Simulated

One or two fuel pins and associated coolant channels.

Modeling

1. The fuel rod or rods and associated coolant channels are divided into radial and axial increments.
2. Considers two-phase convection heat transfer in the slightly subcooled boiling, high subcooled boiling, bulk boiling, and film boiling regimes. Two-phase pressure drop is also considered. Critical heat flux is calculated by several correlations.
3. The conduction heat transfer within the fuel rod is one-dimensional, and heat is generated uniformly in the fuel.
4. The hydraulic model assumes the flow is constant over the length of the channel during each time-step (i.e., steady-state solution

of energy, mass, and momentum balances at each time-step). These assumptions imply that this model is only accurate for slow transients.

5. The two fuel rod models calculate both average fuel pin and coolant channel and hot pin and channel behavior.
6. Requires the fuel pin heat, entering enthalpy, and either inlet flow or pressure drop as input to the code.

Limitations

1. Allows a maximum of 20 radial segments, and the fuel rod axial segment should be no longer than 4 inches.
2. The transients that can be considered must be relatively slow.
3. The reactor coolant must be water at a temperature of less than 1,500°F and between pressures of 150 to 3,000 psi. Fluid qualities up to 40% are considered.
4. A channel must contain at least one fuel rod. There can be no unheated portion of any channel.
5. Metal-water reaction and its effects on heat generation and transfer are not considered.

Computer Operations

1. Operable on the CDC 6600. Written in FORTRAN IV.
2. Run time for a typical case is about 0.6 second per axial node per time-step.

Interface/Overlap

Used in conjunction with other transient analysis codes such as CADD (p. B-19) and PUMP (p. B-93).

Reference

C. D. Morgan, H. C. Cheatword, and J. R. Gloudemans, RADAR - Reactor Thermal and Hydraulic Analysis During Reactor Flow Coastdown, BAW-10069, July 1973.

RELBOT

Purpose

RELBOT calculates the rate of coolant injection from the safety injection tanks (accumulators).

Status

A. Development	Developed by Combustion Engineering Incorporated, Nuclear Power Division, Windsor, Connecticut.
B. Availability	Proprietary.
C. Utilization	Used in conjunction with other LOCA analysis codes to determine the effect on a PWR of a LOCA transient. Has been used for safety analysis report (SAR) calculations at the Palisades Nuclear Generating Station.

Problems Solved

Calculates the rate of safety injection from the safety injection tanks (accumulators) and the total amount of coolant injected.

Systems and Subsystems Simulated

Safety injection tanks (accumulators) and associated piping.

Modeling

1. Includes consideration of piping friction, turning losses, and contraction/expansion losses between the injection tank and the reactor inlet nozzle.

2. The instantaneous pressure difference for injection of coolant is calculated utilizing a table of pressure at the reactor vessel inlet nozzle versus time, which is obtained from an appropriate blowdown code.

3. Assumes an isentropic expansion of the nitrogen cover gas within the safety inspection tank to calculate pressure, i.e., PV^N = constant, $N = 1.4$.

4. Considers the volume of liquid within the piping between the check valve and the safety injection tank.

Limitations

1. Fluid contained in the piping between the check valve and the reactor inlet piping is ignored.
2. The isentropic expansion process may not strictly hold due to heat transfer effects.
3. The static head of fluid within the injection tank is not considered in the pressure difference calculations.

Computer Operations

Operable on the CDC 6600. Written in FORTRAN IV.

Reference

Description of Loss of Coolant Calculational Procedures, CENPD-26, Combustion Engineering Incorporated, NEV Supplements 1, 2, and 3.

[redacted]

SGNIII

Purpose

SGNIII calculates the transient response of the primary coolant system, pressurizer, and steam generators to steam line breaks in order to develop mass/energy release for containment evaluation.

Status

A.	Development	Developed by Combustion Engineering Inc., Nuclear Power Division, Windsor, Connecticut.
B.	Availability	Proprietary.
C.	Utilization	Has been used to generate steam line break mass/energy release data for containment design for many plants in addition to System 80 type plants.

Problems Solved

Determines primary and secondary system thermal and hydraulic coolant conditions along with the core material thermal response to secondary steam line breaks.

Systems and Subsystems Simulated

Reactor primary and secondary systems, including the pressurizer and control systems.

Modeling

1. Consists of fluid flow and heat transfer representations of the reactor core, hot leg plena and piping, pressurizer, steam generators, cold leg piping and plena, and steam generator secondary side by control volumes and flow paths. The steam lines to the turbines are also modeled.
2. Reverse heat transfer from the intact steam generators to the reactor coolant loop is considered.
3. A two-node core fuel model is used together with a point kinetics representation and nonlinear moderator, Doppler, boron, and control element assembly (CEA) reactivity effects.
4. Time-dependent reactivity and coolant flow functions may be input into the code.

5. The secondary side of each steam generator is represented by a finite nodal quasi-static balance of energy flux and secondary fluid thermodynamics.

6. The control systems represented are feedwater flow, turbine bypass, turbine admission valve, pressurizer pressure and level, CEA movement, primary and secondary relief valves, main steam line isolation valves, multiple charging pumps and letdown orifices, and pressurizer spray and heaters.

7. A full range of superheated steam, saturated steam, and water and subcooled water thermodynamic states are available in the pressurizer model.

8. In the simulation, the mathematical equations are in finite difference form and are numerically integrated in time.

9. Although the code emphasis is on accident analysis, it is also capable of predicting the combined response of the system to normal plant load maneuvering transients.

Limitations

1. The reactor primary system is modeled with a core node, a hot leg pipe node, a hot leg plenum node, a cold leg pipe node, a cold leg plenum node, and a steam generator tube node.

2. Each steam generator secondary side and the pressurizer consist of a steam volume and a water volume.

3. The fuel is modeled by two equal volume nodes: one representing fuel and the other representing fuel, gap, and cladding.

Computer Operations

Operable on the Cyber 70/Model 76. Written in FORTRAN IV.

Interface/Overlap

CESEC (p. B-23).

Reference

Preliminary Safety Analysis Report (PSAR), System 80. "Description of the SGNIII Digital Computer Code in Developing Main Steam Line Break Mass/Energy Release Data for Containment Analysis," Appendix 6B, Amendment No. 1, Combustion Engineering Inc., January 1974.

[redacted]

SIRENE

Purpose

SIRENE is designed to facilitate analysis of abnormal transients in a PWR.

Status

A.	Development	Development by CEA Centre d'Etudes Nuclearies de Saclay, Gif-sur-Yvette, France.
B.	Availability	Operational and under further development. Available through the U.S. NRC Light Water Reactor Technical Exchange Program.
C.	Utilization	Used to study transient situations such as control rod drop and excessive load increases.

Problems Solved

Determines the thermal and hydraulic transient response of the parameters of a multiloop PWR.

Modeling

1. Models the multiloop PWR plant. Symmetry allows only one loop to be calculated, if possible.

2. Inputs require only the physical and geometric characteristics of the components and elements.

3. Includes

   a. Core thermal hydraulics and neutron kinetics represented by an average fuel rod and channel. Calculations of the fuel rod temperature are also performed.
   b. Primary system, pipes and tubing, pressurizer, and primary pumps.
   c. Secondary system with once-through U-tube steam generator, steam line, turbine input, and turbine.
   d. Control system including core power imbalance and temperature controls and pressurizer spray, level, and heater controls.

Limitations

1. No limitations on the size of the problem, including geometric modeling.

[redacted]

2. The conservation of mass equation supposes that the fluid is incompressible and, therefore, does not account for local temperature and pressure variations on the flux spatial distribution.
3. Steam and liquid phases cannot coexist in the average channel, and no two-phase flow is considered.
4. Would not work well for situations in which depressurization occurs or for LOCAs.

Computer Operations

Information not readily available.

Reference

J. P. Merle, Adaptation of the SIRENE Code to Study Accidental Transients in Pressurized Water Reactors (PWR), FRRSR-8, UC-78, NRC-4, May 1975.

[redacted]

TOPS

Purpose

TOPS provides a digital simulation of pressurizer dynamics.

Status

A.	Development	Developed by the Westinghouse Electric Corporation at the Bettis Atomic Power Laboratory, West Mifflin, Pennsylvania.
B.	Availability	Released to the National Energy Software Center (NESC) in 1968. Restricted to use by U.S. citizens on either official U.S. Government or commercial nonmilitary applications at a U.S. installation. Cost $61.00.
C.	Utilization	Used to study the thermodynamic paths of pressurizer transients and as a design tool.

Problems Solved

Determines the dynamic response of pressurizer transients based on a rigorous application of the first law of thermodynamics and the laws of phenomenological heat and mass transfer.

Systems and Subsystems Simulated

Pressurizers in PWR power plants.

Modeling

1. The equations for conservation of heat and mass, applied to steam and water phases, together with phenomenological heat and mass transfer laws with empirically determined coefficients are solved by a forward difference integration technique.
2. No prior assumptions are made concerning the particular thermodynamic process followed during in-surges and out-surges. The program itself determines the thermodynamic process at each instant of the transient.
3. It is possible to simulate the special cases of isentropic and saturation line paths by appropriate selection of the heat and mass transfer coefficients.

Computer Operations

1. Operable on the CDC 6600 using Scope 2.0. Written in FORTRAN IV.
2. Requires approximately 100,000 (octal) locations of memory.
3. Typical run time is about 20 seconds.

References

J. A. Redfield and S. G. Margolis, TOPS - A FORTRAN Program for the Transient Thermodynamics of Pressurizers. WAPD-TM-545, December 1965.

C. J. Pfeifer, CDC 6600 FORTRAN Programming, Bettis Environmental Report, WAPD-TM-668, January 1968.

[redacted]

TRANS-FUGUE1

Purpose

TRANS-FUGUE1 performs a transient analysis of a core single-channel, two-phase flow and heat transfer during a postulated reactor incident involving boiling.

Status

A.	Development	Developed by Atomics International, Canoga Park, California.
B.	Availability	Released to the National Energy Software Center (NESC) in 1967. Restricted to U.S. Government contract use only. Cost $61.00.
C.	Utilization	Has been used for core single-channel analysis during postulated reactor power excursions, loss of flow, loss of pressure, and channel plugging incidence.

Problems Solved

Calculates the axial distribution of velocity, void fraction, pressure, coolant temperature, and fuel temperature for a single core flow channel with a specified pitch geometry.

Systems and Subsystems Simulated

Single core flow channel.

Modeling

1. Uses a homogeneous hydrodynamic model that assumes equal phase velocities and phase temperatures.
2. The heat balance equation is solved by use of finite difference techniques, and the momentum equation is derived and solved.

Limitations

1. Allows a maximum of 100 axial nodes.
2. The momentum equation is derived assuming constant vapor density and saturation temperature, eliminating pressure feedback. Numerical instabilities resulted when attempts were made to eliminate these limitations.

3. Homogeneous flow and interphase energy transfer are assumed invalid for separated flow problems.

Computer Operations

1. Operable on the IBM 7094 under the standard FORTRAN monitor. Written in FORTRAN IV.

2. Requires 32,000 bytes of memory and batch input.

3. Additional devices required include an SC-4020 graphical display device with AICRT3 and NAA SC-4020 package software.

4. A typical 80-node problem requires 2 to 4 minutes of computer time.

Reference

R. C. Noyes, J. G. Morgan, and H. H. Cappel, TRANS-FUGUE1, A Digital Code for Transient Two-Phase Flow and Heat Transfer, NAA-SR 11008, July 1965.

Contacts

R. C. Noyes, J. G. Morgan, and H. H. Cappel
Atomics International
P.O. Box 309
Canoga Park, California

Additional software can be obtained through:

Vaide Librarian
c/o Stromberg-Carlson
P.O. Box 2449
San Diego, California

[redacted]

UTSG

Purpose

UTSG calculates the dynamic behavior of a vertical natural-circulation U-tube steam generator together with its steam removal system.

Status

A.	Development	In the development and verification stages by Gesellschaft für Reaktorsicherheit (GRS), m.b.h, F. R. Germany.
B.	Availability	Proprietary.
C.	Utilization	Has been used for post-calculations of start-up tests at PWR nuclear power plants, simulating perturbations both on the primary and the secondary side of the steam generator and for similar calculations for corresponding anticipated transients without scram (ATWS). Comparisons of calculations with start-up test data were favorable.

Problems Solved

Calculates the steam generator and steam removal system thermal-hydraulics with large perturbations in inlet water temperature, inlet water mass flow, and system pressure on the primary side. Feedwater temperature, feedwater mass flow, and outlet steam mass flow disturbed by action of different valves within the steam removal system on the secondary side are also determined. Natural circulation parameters can also be calculated.

Systems and Subsystems Simulated

Vertical U-tube steam generator primary and secondary sides.

Modeling

1. Solves the fundamental conservation and Fourier heat transfer equations.

2. Employs a nodal, nonlinear transient model.

3. Approximations for thermodynamic properties of steam and water, and one- and two-phase flow, heat transfer coefficients, and friction coefficients are incorporated into the code.

Limitations

1. Allows a maximum of seven axial hydrodynamic nodes and three radial heat conduction nodes.
2. For steady-state, a maximum of 91 variables is allowed. For transient, a maximum of 78 variables is allowed.
3. The environmental properties are invalid below 500 psi (saturation).

Computer Operations

1. Operable on the IBM 360 computer. Written in FORTRAN IV.
2. Requires 200,000 bytes of computer-storage capacity.

Interface/Overlap

ALMOD (p. B-7) uses the UTSG code for steam generator and secondary side calculations. TUMHTC, a code which calculates the heat transfer coefficients for nuclear steam generators, performs similar calculations.

References

A. Hoeld, "A Theoretical Model for the Calculation of Large Transients in Nuclear Natural-Convection U-tube Steam Generators," Nuclear Engineering and Design, 47 (1978), pp. 1-23. [Code description.]

W. Frish and K. D. Schmidt, "A PWR Plant Model for the Analysis of Large Amplitude Transients," 3rd Power Plant Dynamics Control and Testing Symposium, Knoxville, Tennessee, September 7 to 9, 1977.

A. Hoeld, "Heat Transfer Coefficients for Nuclear Steam Generators (Digital Code TUMHTC)," GRS-Munich, to be published.

APPENDIX C

Miscellaneous Codes

Section 6 contains miscellaneous codes used for nontransient reactor analysis; these include steady-state analysis codes, system performance codes, and probabilistic and reliability analysis codes.

ALLCUTS

Purpose

ALLCUTS analyzes fault trees for nonminimal and minimal cut sets and optionally determines the set probabilities.

Status

A.	Development	Developed by the Hanford Engineering Development Laboratory, Advanced Waste Engineering Department at the Atlantic Richfield Hanford Company, Richland, Washington.
B.	Availability	Operational. Soon to be released to the National Energy Software Center (NESC).
C.	Utilization	Used to determine the reliability of reactor subsystems.

Problems Solved

Determines nonminimal and minimal cut sets and determines the set probabilities. Can also calculate the maximum-likelihood set duration times.

Modeling

1. Employs an efficient top-down set expansion algorithm to find the cut sets.

2. The major characteristics are

   a. Provides comprehensive error checking of the input data file,
   b. Finds nonminimal or minimal cut sets,
   c. Finds cut sets within any size range from 1 to 10 events,
   d. Finds cut sets within any specified probability range,
   e. Internally stores, orders by probability, and lists up to 1,000 cut sets,
   f. Optionally prints all cut sets found, without storing them internally,
   g. Probability calculations (and therefore input of bottom-event probability data) are optional,
   h. Probability calculations employ the methodology employed in the Rasmussen Report, WASH-1400,
   i. Cut-set listings are printed with bottom-event descriptions at the option of the user,
   j. An optional table of bottom-event importances and importance factors may be printed,
   k. Looping detection and diagnosis is powerful and automatic,

l. Recapitulation of input data is optional,
m. Analysis may start at the top event of the tree or at any intermediate gate,
n. An optional trace printout may be ordered for all paths found from the top of the tree to any gate,
o. All cut sets containing any one bottom event may be listed,
p. Cumulative cut-set probabilities may be requested by cut-set size group and/or for the system as a whole,
q. Performs optional output of cut-set data to a data file for further processing by other codes, e.g., for evaluating cut-set probability spectrums.

Limitations

1. Presently limited to trees with 175 bottom events and 425 gates.

2. Capable of finding cut sets for up to 10 events.

3. These limitations can be changed by the user, depending upon need and computer capability.

Computer Operations

1. Operable on the Cyber 70 computer. Written in FORTRAN IV.

2. Requires 35,000 (octal) words of internal memory storage.

References

W. J. Van Slyke and O. E. Griffing, ALLCUTS, A Fast, Comprehensive Fault Tree Analysis Code, ARH-ST-112, VC-32,41,78, July 1975.

"Reactor Safety Study, An Assessment of Accident Risk in U.S. Commercial Nuclear Power Plants," WASH-1400, (Washington, D.C.: USAEC, to be published). [Draft.]

COMCAN

Purpose

COMCAN is a common cause analysis code that analyzes fault tree minimal cut sets for single secondary events that could cause the occurrence of each event in a minimal cut set.

Status

A.	Development	Developed by EG&G Idaho Inc. at the Idaho National Engineering Laboratory, Idaho Falls, Idaho. COMCAN II is a developmental version.
B.	Availability	Released to the National Software Information Center (NSIC) in 1976. Cost $70.00 (IBM).
C.	Utilization	Currently being tested and evaluated.

Problems Solved

1. Uses input minimal cut sets obtained from other reliability analysis codes. Secondary event susceptibility for those basic events appearing in the minimal cut sets must be supplied.

2. Examines the events in each minimal cut set for common cause susceptibility to generic secondary causes.

3. Location information can also be input by the user. This information is used to locate minimal cut sets containing basic events.

Modeling

1. Uses common cause failure analysis methodology.

2. The COMCAN II code separately examines branches of the tree to locate minimal cut sets, of any order, which are susceptible to single secondary events.

Limitations

1. COMCAN is limited to examining fault tree minimal cut sets obtained from other qualitative analysis codes.

2. The output is usually limited to cut sets of the order three or less.

COMCAN II does not have these two preceding limitations.

Computer Operations

1. COMCAN is operational on the IBM 360 and 370 and the CDC Cyber 76. COMCAN II is operational on the CDC Cyber 76. Written in FORTRAN IV.
2. Input is in batch mode and has dynamic storage capabilities.
3. Requires some form of external storage and the Idaho National Engineering Laboratory (INEL) environment package for dynamic storage.
4. Typical run time for the sample problem (40 minimal cut sets) was 0.4 second on the Cyber 76 and 76 seconds on the IBM 360/75.

Future Efforts

Conversion of COMCAN II to the IBM 360/75.

Interface/Overlap

Requires minimal cut sets from other reliability analysis codes.

References

G. R. Burdick, N. H. Marshall, and J. R. Wilson, COMCAN - A Computer Program for Common Cause Analysis, ANCR-1314 (1976).

D. M. Rasmusson and N. H. Marshall, Preliminary User's Guide to the Reliability Analysis System, RE-S-76-177 (Idaho Falls: Aerojet Nuclear Corporation, September 1976). [Available from ERDA-RDD.]

ELRAFT

Purpose

ELRAFT formulates the simplest logic expression for secondary events in a fault tree in terms of the basic events which combine to cause it.

Status

A.	Development	Developed by Westinghouse Electric Corporation, Nuclear Energy Systems Division, Pittsburg, Pennsylvania.
B.	Availability	Operational.
C.	Utilization	Used to determine event probabilities and failure event combinations for fault trees.

Problems Solved

Calculates event probabilities and lists the combinations of events causing failure arranged by single failure, then double, triple, etc.

Modeling

1. The code uses Boolean algebra to calculate exact probabilities directly from the logic expressions.

2. The probabilities of a secondary event in the fault tree are obtained by

   a. Determining the possible combinations of basic events which cause the secondary event,
   b. Eliminating duplication of events within each combination, and
   c. Eliminating any combination which contains within itself some other possible combination (i.e., ABC contains AB thus ABC must be eliminated).

   The combinations that are left are, for all intents and purposes, mutually exclusive and the probability of the event is the sum of the probabilities of each combination.

3. The property of prime numbers as expressed in the unique factorization theorem of number theory is utilized to eliminate dependencies:

   a. Unique Factorization Theorem--Every natural number greater than 1 can be expressed as a product of prime factors in one and only one way, apart from the order in which the factors are written.

A natural number is called "prime" if it is greater than 1 and has no factors except itself and 1. By assigning a prime number to each basic event, a particular combination of basic events can be expressed uniquely as a single number that is equal to the product of the prime numbers corresponding to the basic events. From this single number, the basic events that make up the combination can be determined by factoring the number into its prime factors.

Limitations

The existing program is designed to handle fault trees which only use AND and OR logic gates. It could, however, be extended to handle any type of logic gate for which there is a mathematical probability equivalence.

Computer Operations

1. Operable on the CDC 6600 using Scope. Written in FORTRAN IV.
2. Requires a minimum of 100,000 (octal) memory.
3. For 100 to 600 total events, run time is about 72 seconds.

Reference

S. N. Semanderes, "ELFRAFT - A Computer Program for the Efficient Logic Reduction Analysis of Fault Trees," IEEE TRANS on Nuclear Science, NS-18:1, February 1971.

Contact

Stavios N. Semanderes
Nuclear Energy Systems Division
Westinghouse Electric Corporation
P.O. Box 355
Pittsburgh, Pennsylvania 15230

FATRAM

Purpose

FATRAM (FAult Tree Reduction AlgorithM) determines minimum cut sets for a fault tree which makes more effective use of computer memory.

Status

A.	Development	Developed by EG&G Idaho, Inc., at the Idaho National Engineering Laboratory, Idaho Falls, Idaho.
B.	Availability	Part of the Reliability Analysis System. Can be obtained from the National Energy Software Center (NESC).
C.	Utilization	Used for system probabilistic reliability analysis.

Problems Solved

Determines the minimum cut set for the top event of a fault tree.

Modeling

1. A top-down algorithm is used.

2. Provides for coding of OR gates so that gate inputs and basic event inputs are separate. This reduces the amount of computer core required.

3. The steps of the alogorithm are as follows:

    a. Resolution begins with the top event. If the top event is an AND gate, all inputs are listed as one set; if it is an OR gate, the inputs are listed as separate sets.

    b. Iterate until all OR gates with gate inputs and all AND gates are resolved. OR gates with only basic event inputs are not resolved at this time.

    c. Any supersets that still exist.

    d. Process any repeated basic events remaining in the unresolved OR gates. For each repeated event,

        - The repeated event replaces all unresolved gates of which it is an input to form new sets,
        - These new sets are added to the collection,

- This event is removed as an input from the appropriate gates, and
- Supersets are removed.

e. Resolve the remaining OR gates. All sets are minimum cut sets.

Limitations

Handles only AND and OR gates.

Computer Operations

1. Operational on the CDC Cyber 76 computer using Scope. Written in FORTRAN IV.

2. Requires a minimum of 25,000 (octal) words of central memory. No external storage capacity is necessary.

3. A program with 306 gates, 418 basic events, and 87 repeated events with a maximum order of cut sets of 2 required 2.3 seconds of computer run time.

Interface/Overlap

ELRAFT (p. C-7), MOCUS (p. C-43), MISCUP, SETS (p. C-67), and ALLCUTS (p. C-3) perform similar calculations.

References

D. M. Rasmusson and N. H. Marshall, "FATRAM - A Core Efficient Cut Set Algorithm," IEEE Transactions on Reliability, Vol. R-27, No. 4, October 1978.

D. M. Rasmusson, N. H. Marshall, and G. R. Burdick, User's Guide for the Reliability Analysis System (RAS). TREE-1168, September 1977.

Contacts

Dr. D. M. Rasmusson
EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, Idaho 83401

N. H. Marshall
EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, Idaho 83401

FAULTRAN

Purpose

FAULTRAN performs analysis of fault trees using bit manipulation and determines minimum cut sets and top-event probabilities.

Status

A.	Development	Developed by Corporate Research and Development, General Electric Company, Schenectady, New York.
B.	Availability	Operational.
C.	Utilization	Has been tested with fault trees of several sizes.

Problems Solved

Generates minimum cut sets for fault trees and determines the top-event existence probability with upper and lower bounds for S-independent minimum cut sets composed of S-independent basic events.

Modeling

1. Uses binary coding of events and a bit-manipulating algorithm to determine minimal cut sets.

2. The output of lower-level gates is used as input to higher level AND or OR gates. The output of an AND gate is the bit-by-bit logical sum of the inputs which can be obtained with the intrinsic OR function. The output of an OR gate is the list of nonduplicated inputs.

3. The probability of the top event is expressed in series form with each term being formed by continuing the binary representation of the minimal cut sets and using the OR function to generate the desired intersection of cuts. The process is often terminated at the first- and second-order terms in the series, thereby producing upper and lower bounds.

4. Includes an error detection capability that can check whether the gate type is AND or OR, check that the event probability is between 0 and 1, and check for inconsistencies in the tree structure description.

Limitations

1. Presently scaled for 100 primary events for the interactive version and 500 primary events for the batch version. This limit can

vary, and problem size is only limited to the memory of the computer employed.

2. Only fault trees with AND and OR gates are analyzed.

Computer Operations

1. Operable on the Honeywell 600/6000 computer. Written in FORTRAN IV.

2. Can be used in either batch or interactive mode. Batch mode is input from data files, and the interactive mode asks pertinent questions.

3. The computation time for a problem with 68 events, 85 gates, and producing 448 minimal cut sets was about 60 seconds on the Honeywell 635 with batch input.

Interface/Overlap

ELRAFT (p. C-7) performs similar calculations.

References

D. B. Wheeler, J. S. Hsuan, R. R. Duersch, and G. M. Roe, "Fault Tree Analysis Using Bit Manipulation," IEEE Transactions of Reliability, Vol. R-26, No. 2 (June 1977).

S. N. Semanderes, "ELRAFT, a Computer Program for the Efficient Logic Reduction of Fault Trees," Trans. American Nuclear Society, Vol. 13, June 1970, pp. 781-782.

Contact

D. B. Wheeler
Corporate Research and Development
General Electric Company
Schenectady, New York 12345

FAUNET

Purpose

FAUNET is a fault tree and network evaluation program.

Status

A.	Development	Developed by the Electronics Department of the Research Establishment, Risø, Denmark.
B.	Availability	Operational.
C.	Utilization	Used for the reliability and safety analysis of complex systems. Has been applied to the ac emergency power system of nuclear power plants.

Problems Solved

Determines the minimal cut/path set of a tree and the unavailability of a system at various time points.

Modeling

1. Provides a graphical representation from the input data so that a tree can be easily surveyed.
2. Performs various error checks on the input data to see if the tree is connected and if there are multiple definitions of a gate.
3. Makes a top-down calculation of the minimal cut/path sets. A pruning of the fault tree may be performed before the minimal cut set calculation.
4. Reliability and availability calculations are performed from minimal sets and repair and test data for the components in the system.
5. Expects failure probabilities and repair times which are either constant or exponentially distributed. Specification of test intervals is also allowed.
6. For networks in which links (components) and/or nodes can fail, the program calculates all paths between two specific nodes. The program can handle networks with both uni- and bi-directional links.

Limitations

1. Limited to 1,000 events and 1,000 gates. Exceptionally long and complicated trees or networks may overflow the computer for which this program was developed.

2. Only AND and OR gates are considered with a primal or dual tree.

Computer Operations

1. Operable on the PDP-8 minicomputer using OS/8. Written in standard FORTRAN. (With minor modification, the program can be implemented on any computer that accepts FORTRAN.)

2. Requires 8,000 bytes of memory.

Reference

O. Platz and J. V. Olsen, FAUNET: A Program Package for Evaluation of Fault Trees and Networks, Risø Report Number 348, September 1976.

Contact

Jul Gjellerup, Sales Distribution
87, Sølvgade, DK-1307
Copenhagen K, Denmark

Available on exchange from
Risø Library
Research Establishment
Risø, DK-4000 Roskelde, Denmark

FRANTIC-NRC

Purpose

FRANTIC (Formal Reliability Analysis including Normal Testing, Inspection and Checking) was developed to investigate periodic testing schemes and operational and design modifications as they affect system unavailability and accident probabilities.

Status

A.	Development	Developed by the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, D.C.
B.	Availability	Released to the National Energy Software Center (NESC) in 1977. Cost $61.00.
C.	Utilization	Used to analyze various testing schemes, operational changes, and design modifications in large complex systems for their effect on system reliability. Represents an extension of the probabilistic methodology described in WASH-1400.

Problems Solved

1. Calculates both the instantaneous and average unavailabilities for any general system and gives a breakdown of the unavailability contributions from failures, testing, and repair.

2. The time-dependent system unavailability function is output in the form of a table or a graph.

Modeling

1. Provides a detailed, time-dependent modeling of periodic testing that includes effects of test downtimes, test overrides, detection inefficiencies, and test-caused failures.

2. Accident sequences, such as those constructed from event trees, can be evaluated for their instantaneous and average probability behavior.

3. Nonrepairable components, monitored components, and periodic testing components are considered.

4. The exponential failure distribution is used for hardware failures. The constant failure rates can be changed to determine the effects of different hardware reliabilities. Periodic equations determine the testing and repair contributions. Human error and

common mode failures can be included by assigning an appropriate constant probability for the contributors.

5. Testing characteristics which can be input include the test interval, test duration time, repair time and allowed downtime, the test override capability, test efficiency, and associated human-caused failure probabilities.

Computer Operations

1. Operable on the IBM 370 using OS/370. Written in FORTRAN IV.
2. Requires 140,000 bytes of memory for execution.
3. Uses proprietary CALCOMP software and system-dependent routines to produce graphical output. However, an alternative GRAPH subroutine is provided for printer plots rather than CALCOMP plots. This GRAPH subroutine requires the NIH IPP plot package.
4. Run time is dependent upon the complexity of the problem. The sample problem required 10 CPU seconds on the IBM 370/195.

References

W. E. Vesely and F. F. Goldberg, FRANTIC - A Computer Program for Time-Dependent Unavailability Analysis, NUREG-0193, October 1977.

W. E. Vesely and F. F. Goldberg, "Time-Dependent Unavailability Analysis of Nuclear Safety Systems," Institute of Electrical and Electronic Engineers Transactions on Reliability, Volume R-26 (October 1977), pp. 257-260.

Reactor Safety Study - An Assessment of Accident Risk in U.S. Commercial Nuclear Power Plants, WASH-1400, NUREG-75/014, October 1975.

FTA

Purpose

FTA was designed to predict failure probabilities for complex systems and to represent graphically the structure of the systems.

Status

A.	Development	Developed by Atlantic Richfield Hanford Company, Richland, Washington, and Computer Sciences Corporation, Richland, Washington.
B.	Availability	Released to the National Energy Software Center (NESC) in 1975. Cost $66.00.
C.	Utilization	Used to determine the failure probabilities of complex systems.

Problems Solved

Determines the cut sets and probabilities of the modes of failure for complex systems and graphically represents the fault tree structure of the systems.

Modeling

1. FTA consists of three programs:
    a. ALLCUTS performs the cut set and probability calculation,
    b. KILMER constructs a CALCOMP plot of the system fault tree, and
    c. BRANCH builds a cross-reference list of the system fault tree.
2. ALLCUTS employs a top-down set expansion technique to find the fault tree cut sets and optionally calculate their probabilities.
3. KILMER is a interactive program which builds a plot file according to user-supplied instructions.

Limitations

1. Allows a maximum of 175 basic events and 425 rate events.
2. May be expanded to solve larger problems depending upon available computer memory.

Computer Operations

1. Operable on the CDC 6600 computer using Scope 3.4.2. Written in FORTRAN IV (95%) and Compass(5%).

2. ALLCUTS requires 110,000 (octal) words, KILMER requires 65,000 (octal) words, and BRANCH requires 50,000 (octal) words of computer memory.

3. Run time is dependent upon the problem size and plotting requirements. Typical times are from 20 to 50 seconds.

References

J. Diven, D. Griffing, A. Thorpe, and W. J. Van Slyke, Fault Tree Analysis System Users Guide, CSC-RL-SCI-122, June 1975.

W. J. Van Slyke and D. Griffing, ALLCUTS - A Fast Comprehensive Fault Tree Analysis Code, ARH-ST-112, July 1975.

GO

Purpose

GO was developed to analyze event trees and fault trees and to determine probabilities and cut sets.

Status

A.	Development	Developed by Kaman Sciences Corporation for the Electric Power Research Institute, Palo Alto, California.
B.	Availability	Proprietary.
C.	Utilization	Developed to analyze the safety and reliability of complex electromechanical systems.

Problems Solved

1. Determines the possible system response modes, both successes and failures, and their probabilities for event trees.

2. Also determines nonminimal and minimal cut sets for a selected event and their probabilities for fault trees.

Modeling

1. Five specific capabilities are provided:

   a. Ease of system characterization using standardized and generalized components,
   b. Inclusion of time and sequence operations,
   c. The simultaneous generation of the entire set of system operational modes,
   d. The technique of combining events which differ only in details that are no longer relevant (branch combining), and
   e. The selective pruning of events based on the magnitude of the probability of occurrence of the event combinations as they are generated (the threshold for pruning is user specified).

2. Each component in the system is modeled by specifying the type of component and the probabilities of the subsequent operations. (The component type is determined according to the standardized manner in which input and output are handled.)

3. The code consists of 16 standardized types of component operations

   a. Two-state component,
   b. Normally open contact,

c. AND gate,
d. OR gate,
e. Linear combination generator,
f. Normally closed contact,
g. Value/probability gate,
h. Triggered generator,
i. Delay generator,
j. Path sputter,
k. Actuated normally closed contact,
l. Actuated normally open contact,
m. Function operator,
n. Multiple input/output operator,
o. Multiple input/single output operator (M out of N gate), and
p. Signal operator.

4. Fault tree analysis is performed for a specified event by indicated AND and OR gates for the tree.

Limitations

1. The maximum number of signals allowed is 9,999; the identifying signal number must be unique for each signal.

2. The number of the signal value must be between 2 and 128. (The signal is the random variable.)

3. The maximum number of final signals allowed is 24.

4. The maximum number of signal inputs and outputs for a regular multiple output generator is 10.

5. The specific parameters for a given operator are limited to 100.

Computer Operations

1. Developed for the CDC 6400 using the FTN computer on the NOS/BE-L operating system.

2. The memory size necessary is dependent upon program length and complexity.

3. Run time for an example event tree analysis with 35 operators producing 5 final signals was approximately 25 seconds.

References

GO Methodology - System Reliability Assessment and Computer Code Manual, EPRI NP-766, Project 818 (Palo Alto: Electric Power Research Institute, May 1978).

GO Methodology - Fault Sequence Identification and Computer Code Manual, EPRI NP-767, Project 818 (Palo Alto: Electric Power Research Institute, May 1978).

Contact

Kaman Sciences Corporation
3412 Hillview Avenue
Palo Alto, California 94304

[redacted]

HAFMAT

Purpose

HAFMAT determines the steady-state distribution of flow to all paths of any given flow system. The system is considered as a network of flow paths or pipes.

Status

A.	Development	Developed by the General Electric Company at the Knolls Atomic Power Laboratory, Schenectady, New York.
B.	Availability	Released to the National Energy Software Center (NESC) in 1975. Restricted to use by U.S. citizens for either official U.S. Government or commercial nonmilitary applications at a U.S. installation. Cost $64.00.
C.	Utilization	Used to analyze steady-state plant performance and design studies.

Problems Solved

Determines steady-state flow distribution, pressure drops, coolant enthalpies, and heat addition to the coolant.

Modeling

1. The flow geometry is represented by a finite number of flow paths which account for all of the fluid volume. These flow paths are joined at a number of points, or nodes (junctions), to form a single flow network.

2. The nodes or junctions do not have any capacity for heat or flow.

3. A matrix technique is used to determine the static pressure at each node and a new self-consistent flow distribution in the paths. The program then iterates through the new flow distribution to obtain new values for enthalpy and pressure drop until convergence is achieved or an iteration limit is reached.

4. The basic assumptions employed are that complete and instantaneous mixing of fluid occurs at all nodes and all flows which join at a node have the same static pressure at that node.

5. The user may specify the input pressure drop equation if the preprogrammed ones are not appropriate.

Limitations

A maximum of 90 internal nodes with 200 internal branches, 20 external nodes, 20 external branches with 100 heated branches, and 40 types of heated branches are allowed.

Computer Operations

1. Operable on the CDC 6600,7600 using Scope 3.2. Written in FORTRAN IV.
2. Requires 76,000 (octal) words of core memory.
3. The sample problem required approximately 5 seconds of CPU time.

Interface/Overlap

FLOPSY-I, the original version of HAFMAT, treated only isothermal flow problems.

Reference

L. H. Wunderlich and D. R. Dolk, HAFMAT - Steady-State Flow Distribution Program, KAPL-M-7128 (LXW-2), July 1970.

[redacted]

HAMBO

Purpose

HAMBO was developed for the steady-state hydraulic and burn-out analysis of a rod-cluster assembly cooled at a boiling fluid using a subchannel model.

Status

A. Development	Developed by the Steam Generating Heavy Water Reactor Development Division of the Atomic Energy Establishment, Winfrith, Dorchester, Dorset, England.
B. Availability	Operational.
C. Utilization	Used primarily as a design tool for thermal performance assessment of fuel assemblies in water/steam cooled nuclear reactors and for burn-out assessment in rod-clusters.

Problems Solved

1. Determines the effects of changes in cluster radial flux depression, rod pitch, bowing, and eccentricity on fluid density and burn-out.
2. The solution gives the mass velocity, enthalpy, pressure drop, vapor quality, void fraction, and mean fluid density for each subchannel.

Systems and Subsystems Simulated

Nuclear rod-cluster assembly and associated coolant subchannels.

Modeling

1. The rod-cluster assembly is sectioned into subchannels, and the program divides the channel length into a number of calculation intervals.
2. The thermal and hydraulic equations are solved by a finite difference method for each interval, which allows a convenient representation of nonuniform axial heat flux.
3. The subchannel equations of mass, heat, and momentum balance over each interval are solved by an iterative process which employs as a pressure balance criterion that the pressure difference between subchannels should correspond to the lateral mass transfer.

4. The following physical effects are included in the hydraulic model:

    a. Heat and momentum transfer between adjacent subchannels by mass transfer (cross-flow) and by turbulent mixing,
    b. Single-phase, subcooled boiling and bulk boiling regimes,
    c. Rod spacing by localized grids or wire wraps,
    d. Radial and axial heat profiles, and
    e. Lateral resistance to cross-flow.

5. The subchannel input data include details of the area, heat and wetted parameters, and power distribution. The boundaries between adjacent subchannels are also specified, together with the geometrical shape of the boundary.

6. The following options are available as input data for the hydraulic and burn-out models:

    a. Single-phase friction coefficients and grid pressure less coefficients may be given as arbitrary functions of Reynolds Number,
    b. Two-phase friction is calculated by the "homogeneous theory with slip," Martinelli-Nelson, Thom, or an arbitrary correlation,
    c. Slip ratio is calculated by the Bankoff-Jones correlation or as an arbitrary function of quality,
    d. Physical properties are calculated internally for water over the pressure range from 15 to 2,500 psia or are specified as input data, e.g., for other fluids,
    e. Inlet flow division to subchannels is calculated by pressure balance or specified, together with inlet enthalpy, as input data,
    f. Vertical, inclined, or horizontal flow. (The latter two are, strictly, only applicable to the mixed flow model since the subchannel model does not treat the radial separation of liquid and vapor due to gravity effects),
    g. An electric mix flow correlation,
    h. Westinghouse WCAP (subchannel) correlations,
    i. Westinghouse WAPD 188 round tube and rectangular channel correlations (9),
    j. Macbeth round tube and mixed flow cluster correlations,
    k. The Macbeth-Thompson round tube correlation,
    l. Barnett annuli and mixed flow cluster correlations, and
    m. HAMBO subchannel correlations (unpublished).

Limitations

1. A maximum of 10 subchannels are allowed with up to 37 rods and a maximum of 20 boundaries. These limitations are based on computer size.

2. No account is taken of the change in saturation temperature with change in pressure along a channel, which would cause flashing of some of the superheated water into steam.

[redacted]

3. Resistance to lateral flow is due entirely to friction.

4. Allows no transition region of finite enthalpy and velocity across the subchannel boundaries.

5. The single-phase turbulent mixing model was extended to the two-phase region by means of the homogeneous theory.

6. Two-phase flow patterns are neglected.

Computer Operations

1. Operable on the KDF-9 and the IBM 7090/7030 computers. Written in Egran (KDF-9) and FORTRAN IV (IBM).

2. The code requires 15,500 words on the KDF-9 computer.

3. Typical run time is 2 minutes to read the program plus 15 seconds for a 7-rod cluster, 30 seconds for a 19-rod cluster, or 150 seconds for a 37-rod cluster on the KDF-9 computer.

Interface/Overlap

HAMBO supercedes the earlier subchannel codes LIFEBUOY, BLOWLINE, and TWOSTEP.

References

R. W. Bowring, HAMBO - A Computer Programme for the Subchannel Analysis of the Hydraulic and Burnout Characteristics of the Rod Clusters, Part 1-General Description, AEEW-R524, 1967.

D. Burns et al., "The Egdon System for KDF 9," Computer J., 8, 297 (1966).

[redacted]

HECTIC II

Purpose

HECTIC II determines coolant behavior in heat exchanges such as fuel elements of gas- or liquid-cooled nuclear reactors under steady-state conditions.

Status

A. Development	Developed by Aerojet-General Nucleonics, San Ramon, California, as part of the Army Gas-Cooled Reactor Systems Program.
B. Availability	Operational.
C. Utilization	Used to analyze nuclear fuel element performance and for design calculations. Also used to study a variety of reactor problems associated with turbulent flow.

Problems Solved

Calculates pressure drops, flow rates, heat transfer rates, and coolant temperatures in fuel element coolant channels as a function of distance downstream from the inlet.

Systems and Subsystems Simulated

Nuclear reactor fuel elements and coolant passage for gas- or liquid-cooled reactors.

Modeling

1. The geometry of the coolant passage is modeled by the minimum symmetrical section representative of the reactor fuel element in which several heated surfaces (such as rods and plates) are present.
2. The coolant passage is divided into several flow passages and lumped surfaces. The axial mesh is automatically determined by the program to assure that a prescribed accuracy of results is obtained and that the pressure drop for each axial segment is equal.
3. Widely varying pin and coolant passage geometrics can be considered.

4. Includes turbulent heat and momentum exchange between all considered flow passages along with wall friction, drag against spacers, and acceleration forces.

5. Heat transfer by conduction, convection, and radiation are considered along with ambient heat loss calculations.

6. The computation procedure amounts to a "nodal" or "lumped parameter" type of calculation.

Limitations

1. Allows a maximum of 30 flow passages and 24 surface lumps.

2. Considers only steady-state flow conditions of a single-phase gas or liquid coolant.

3. The flow areas for each axial segment cannot vary (constant cross-sectional area).

4. Turbulent flow is considered only.

Computer Operations

1. Operable on the IBM 7090 computer. Written in FORTRAN.

2. A typical problem requires approximately 0.5 to 5 minutes of computer time, depending upon the number of fuel rods (or plates) and passages and various options incorporated into the program.

Interface/Overlap

HECTIC is an earlier version of HECTIC II.

References

N. Kattchee and W. C. Reynolds, HECTIC II - An IBM 7090 FORTRAN Computer Program for Heat Transfer Analysis of Gas or Liquid Cooled Reactor Passages, IDO-28595, (California: Aerojet-General Nucleonics, revised December 1965).

W. C. Reynolds et al., HECTIC, An IBM 7090 Computer Program for Heat Transfer Analysis of Gas Cooled Reactors, AGN-TM-381, (California: Aerojet-General Neucleonics, May 1961).

[redacted]

IMPORTANCE

Purpose

IMPORTANCE computes the various measures of probabilistic importance of basic events and determines minimal cut sets of a fault tree or reliability network diagram.

Status

A. Development	Developed by the Lawrence Livermore Laboratory, Livermore, California, for the CDC 7600 and converted to the IBM 370 by the Reactor Analysis and Safety Division of the Argonne National Laboratory, Argonne, Illinois.
B. Availability	The IBM 370 version was released to the National Energy Software Center (NESC) in 1978. Cost $61.00.
C. Utilization	Used to quantitatively evaluate reactor systems safety and reliability.

Problems Solved

1. Determines the probability of the top event of a fault tree and computes the importance of the minimal cut sets and basic events.

2. Two measures are computed:

    a. System behavior at one point in time, and
    b. The sequences of failures that cause the system to fail in time.

3. In addition, system unavailability and expected number of system failures are also computed.

Modeling

1. Seven measures of basic event importance and two measures of cut-set importance can be computed. Birnbaum's measure of importance (i.e., the partial derivative) and the probability of the top event are computed using the min cut upper bound. If there are no replicated events in the minimal cut sets, then the min cut upper bound is exact. If basic events are replicated in the minimal cut sets, then based on experience the min cut upper bound is accurate if the probability of the top event is less than 0.1.

2. Simpson's rule is used in computing the time-integrated measures of importance. Newton's method for approximating the roots of an

equation is employed in the options in which the importance measures are computed as a function of the probability of the top event, and a shell sort puts the output in descending order of importance.

3. The minimum cut sets, the failure rate data, and fault duration times (i.e., the repair times) of all the basic events contained in the minimum cut sets are supplied by the user.

4. For systems in which repair is not allowed, the code accepts proportional hazards as input data. In another option, in which repair is permitted, failure rate data can also be expressed in relative terms by representing the failure rate and repair rate data for the basic events in terms of a reference time unit.

Computer Operations

1. The public domain version is operational on the IBM 370 using OS/370. Written in FORTRAN IV.

2. Requires 430,000 bytes of memory for the IBM 370/195 for a typical problem.

3. The test problem executed in 1 second on IBM 370/195.

Interface/Overlap

SETS (p. C-67), MOCUS (p. C-43), FTA (p. C-17), PREP (p. C-49), and COMCAN (p. C-5) perform similar calculations.

References

H. E. Lambert, *Fault Trees for Decision Making in Systems Analysis*, UCRL-5189, 9 October 1975. [Ph.D. thesis.]

Howard E. Lambert and Frank M. Gilman, *The IMPORTANCE Computer Code*, UCRL-79269 (preprint), 14 March 1977.

INTEG/INSPEC

Purpose

The INTEG/INSPEC programs analyze the characteristics of a general model developed to represent the safety aspects of an operating nuclear reactor.

Status

A.	Development	Developed by E.I. Du Pont de Nemours at the Savannah River Laboratory, Aiken, South Carolina.
B.	Availability	Released to the National Energy Software Center (NESC) in 1973. Cost $57.00.
C.	Utilization	Used to determine reliabilities, probabilities, and frequencies of incidents for operating reactor systems.

Problems Solved

Analyzes the characteristics which are the frequencies of incidents that are departures from the expected behavior of the reactor.

Modeling

1. Each incident is assumed to be preceded by a sequence of events starting at some initiating event. At each member in the sequence, there may be functions such as safety circuits and personnel operations that stop the sequence at that number.
2. When mechanical devices fail, they are assumed to remain inoperative until repair.
3. The model accounts for scheduled inspection and maintenance of all equipment in the system.
4. In INTEG, the discontinuous density function is integrated by the trapezoidal rule from time zero to time T.
5. INSPEC is based on the simulation of reactor operation as a Markov process. A vector of probabilities is successively multiplied by a transition matrix.

Limitations

1. In INTEG, failure rates of safety circuits are restricted to be independent.

2. INSPEC is limited to subsystems with no more than seven safety circuits.

3. The transition matrix can be made up as desired so that any intercorrelations between failure circuits can be accommodated.

Computer Operations

1. Operable on the IBM 360 using OS/360. Written in FORTRAN IV.

2. The INTEG sample problem containing five problems, each with 3,000 steps of trapizoidal integration and an average of 4.6 safety circuits, requires a total CPU time of 2 seconds on the IBM 360/195.

3. The INSPEC sample problem with three safety circuits for 500 transitions had an indicated CPU run time of less than 2 seconds.

Reference

L. M. Arnet, Quantitative Analysis of Reactor Safety, DP-1168, November 1968.

LYNX1/LYNX2

Purpose

The LYNX/LYNX2 code package is used to determine the steady-state thermal and hydraulic conditions of the coolant flow in a reactor core.

Status

A.	Development	Developed by Babcock and Wilcox (B&W), Power Generation Group, Nuclear Power Generation Division, Lynchburg, Virginia.
B.	Availability	Proprietary.
C.	Utilization	Used to determine steady-state thermal and hydraulic conditions of the core based on the geometry and operating conditions of a PWR. Has been used to determine departure from nucleate boiling ratio (DNBR).

Problems Solved

1. Determines the detailed fluid and flow conditions in the reactor core.
2. LYNX1 determines the fluid and flow conditions for the fuel bundles in the core with bundle-to-bundle cross-flow.
3. LYNX2 uses the LYNX1 calculations and determines fluid and flow conditions for the subchannels within a fuel bundle accounting for intersubchannel cross-flow and turbulent interchange.

Modeling

1. LYNX1
    a. Each fuel bundle is considered as a lumped unit in the reactor cross-sectional plane, and the core is divided into a finite number of axial increments.

    b. LYNX1 is based on the one-dimensional conservation equations of mass, momentum, and energy formulated in the axial direction; the effects of thermal nonequilibrium and slip flow on the fluid properties are accounted for by empirical models and correlations.

    c. A simplified conservation equation for transverse momentum is used to calculate the interbundle cross-flow.

d. The boundary value problem is solved iteratively with either the core inlet velocity or pressure profile supplied, so that the boundary condition of a given pressure profile at the core exit is satisfied.

2. LYNX2

a. The fuel bundle is modeled with fuel rods and associated flow subchannels and divided into axial increments.

b. Based on the one-dimensional conservation equations for a single- and two-phase fluid. The effects of nonequilibrium and slip flow are accounted for by empirical models and correlations.

c. Interchannel diversion cross-flow is determined from transverse pressure differences. Also considered is intersubchannel turbulent interchange.

d. The coupled conservation relations for the linked subchannels are iterated to a solution over each calculational increment. Interbundle cross-flow from LYNX1 is matched in the two normal flow directions with the periphery rod and subchannel locations.

e. The geometry, location, area and gap variations, and hot-channel-factor set of each subchannel may be described, and the geometry, location, radial power factor, and axial flux shape for each fuel rod may be defined. Form loss regions may be modeled using position, length, loss-coefficient, and area contraction coefficients. The turbulent interchange proportionality factor may be input as a function of quality. Critical heat flux ratios may be calculated using the B&W2 and W-3 correlations.

Limitations

1. LYNX1

a. The maximum number of bundle rows and columns allowed is 9 by 9.

b. The minimum number of bundle rows and columns allowed is 2 by 2. To satisfy this limitation for bundle arrays less than 2 by 2, additional dummy bundles can be added.

c. The maximum number of axial increments allowed is 100.

d. The reactor coolant is limited to water at pressures between 1,000 and 3,208 psia and temperatures less than 1,500°F.

e. Axial flow is limited to positive; no reverse flow is allowed in the core.

2. LYNX2

   a. The subchannel model is limited to an array from 4 to 324 linked subchannels and 1 to 289 fuel rods.

   b. The code allows a maximum of five hot-channel-factor sets and five polynomials or four section sets for axial heat flux shapes.

   c. A maximum of 10 form loss regions may be modeled.

   d. LYNX2 is limited to water at pressures less than 3,208.5 psia and temperatures less than 1,500°F.

Computer Operations

1. Operable on the CDC 7600 using Scope. Written in FORTRAN IV.

2. Typical run time for each of these programs is about 30 seconds.

Interface/Overlap

COBRA (p. A-17) performs similar calculations.

References

B. R. Hao and J. M. Alcron, LYNX1-Reactor Fuel Assembly Thermal Hydraulic Analysis Code, BAW-10129, October 1976.

Power Generation Group, Nuclear Power Generation Division, LYNX2, Subchannel Thermal-Hydraulic Analysis Program, BAW-10130, (Lynchburg: Babcock and Wilcox, October 1976).

MFAULT

Purpose

MFAULT analyzes fault trees and determines cut sets and probabilities.

Status

A.	Development	Developed by Battelle-Columbus Laboratories at the Pacific Northwest Laboratory, Richland, Washington.
B.	Availability	Operational.
C.	Utilization	Used for the analysis of nuclear fuel cycle operations. Has been used for risk analysis of systems for managing high-level radioactive waste.

Problems Solved

Identifies cut sets in fault trees, calculates their probabilities, and screens them on the basis of specified cut-offs on probability and/or cut-set length.

Modeling

1. Based on an efficient upward-working algorithm for cut-set identification. This algorithm estimates the number of binary indicated cut sets versus the number of events per cut set.

2. The cut-set probability calculation is optional. These calculations, when performed, are based on the assumption of small probabilities and constant hazard rates (i.e., exponential failure distribution).

3. Cut sets consisting of repairable components (basic events) only, nonrepairable components only, or a mixture of both component types can be evaluated. Components can be on-line or standby. Unavailability contributions from pre-existing failures, failure on demand, and testing and maintenance downtime can be handled.

4. Has the capability of reading in the complete fault tree, then specifying any gate as the top gate to analyze. Fault trees with AND gates, OR gates, inhibit gates, ON switches (houses), and OFF switches can be analyzed.

5. The program has the following options:

   a. Using a probability cut-off,
   b. Using a cut-off on the number of events per cut set,

c. Recording the disposition of cut sets at each gate,
d. Printing cut sets and/or creating a file,
e. Creating a file of cut sets discarded at the top event,
f. Checking for the redundancy of cut sets, and
g. Finding minimal cut sets.

Limitations

Capable of finding up to 10 event cut sets from a fault tree of 512 basic events.

Computer Operations

1. Operable on the CDC Cyber 74 computer.

2. Run time is a function of tree size, structure, and the value of the cut-offs, if used. Complex fault trees containing millions of cut sets have been evaluated in terms of the top few hundred or thousand cut sets. Computer requirements were on the order of 5 minutes on the Cyber 74 system.

Interface/Overlap

MFAULT can be used with the ACORN code, which draws fault trees, and the RAFT code, which calculates risk.

References

P. J. Pelto and W. L. Purcell, MFAULT - A Computer Program for Analyzing Fault Trees, BNWL-2145, UC-32, November 1977.

J. L. Carter, ACORN - A Computer Program for Plotting Fault Trees, BNWL-2144, November 1977.

G. D. Seybold, RAFT - A Computer Program for Fault Tree Risk Calculations, BNWL-2146, November 1977.

MOCARS

Purpose

The Monte Carlo Simulation (MOCARS) code is used to determine the distribution and simulation limits for a system unavailability function or a user-supplied function.

Status

A.	Development	Developed by EG&G Idaho, Inc., at the Idaho National Engineering Laboratory (INEL), Idaho Falls, Idaho.
B.	Availability	Operational.
C.	Utilization	Currently being tested and evaluated.

Problems Solved

1. Can be used to calculate the distribution and simulation limits either for a system unavailability function using the minimal fault tree cut sets or for a user-supplied function. The fault tree cut sets can either be entered directly via the Reliability Analysis System (RAS) code (p. C-55) or can be input from another fault tree code, such as PREP (p. C-49).

2. The user can also enter any function for which distribution and simulation limits are desired. Any number of input variables can be entered for a MOCARS problem in conjunction with 12 common statistical distributions to evaluate the user function.

Modeling

1. Uses any of 12 statistical distributions to sample data for evaluation of the cut-set system unavailability function or the user-supplied function.

2. These 12 distributions are the following: uniform, exponential, normal, Cauch, gamma, beta, log normal, binomial, Poisson, Weibull, Pearson Type IV, and empirical.

Limitations

Limited to determining the distribution and simulation limits for a system unavailability function using the minimal cut sets or a user-supplied function using the statistical distribution given above. If cut sets are entered, no more than 7 events per cut set can be entered; no input restrictions exist for a user-supplied function input.

Computer Operations

1. Currently operational on the CDC Cyber 76. Written in FORTRAN IV.

2. Operates in batch mode, has dynamic storage capabilities, "free form" input capability, and microfilm graphics capability to plot the frequency and cumulative sample histograms. Requires some form of external storage, such as disk or drum. Requires the use of the INEL environmental package for the dynamic storage capability, the "free form" input, and the Integrated Graphics System (IGS).

3. Typical run time for the cut-set sample problem was approximately 3 seconds to analyze 55 cut sets. For the user-supplied function sample problem, approximately 2 seconds was required. This included link-edit and execution with all options exercised.

Future Efforts

Conversion of MOCARS to the IBM 360/75.

Interface/Overlap

For cut-set input, MOCARS has the capability to interact with RAS (p. C-55) to obtain the minimal cut sets.

References

1. S. D. Matthews, MOCARS: A Monte Carlo Simulation Code for Determining the Distribution and Simulation Limits, RE-A-77-001, May 1979.

2. D. M. Rasmuson and N. H. Marshall, Preliminary User's Guide to the Reliability Analysis System, RE-S-76-177, September 1976.

3. W. E. Vesely and R. E. Narum, PREP and KITT: Computer Codes for the Automatic Evaluation of a Fault Tree, IN-1344, August 1970.

MOCUS

Purpose

MOCUS determines the system failure modes (minimal cut sets) and system success modes (minimal path sets) from a system fault tree.

Status

A.	Development	Developed by the Aerojet Nuclear Corporation at the Idaho National Engineering Laboratory, Idaho Falls, Idaho.
B.	Availability	Released to the National Energy Software Center (NESC) in 1975. Cost $64.00.
C.	Utilization	Has been used to analyze the safety of components and systems in nuclear power plants.

Problems Solved

Determines the requested minimal cut sets and minimal path sets.

Modeling

1. A description of the system fault tree and control parameters specifying the minimal cut-set length to be obtained is required as input.
2. Uses direct resolution, starting with the main feature of interest, which is the top event, and then proceeding to the basic independent component failures, called the primary event.
3. Guarantees that all the minimal sets requested have been found.

Limitations

1. The output can include minimal cut and path sets for a maximum of 20 gates.
2. Gates must be described only in terms of AND and OR logic.
3. An AND gate alone increases the number of path sets and an OR gate alone increases the number of cut sets.

Computer Operations

1. Operable on the IBM 360/370 using OS/360. Written in FORTRAN.

2. Requires 222,000 bytes for execution and 7 input/output (I/O) units.

3. Typical run time is about 2 minutes on the IBM 360/195.

Interface/Overlap

The fault tree description input to MOCUS is identical to the input description in the PREP (p. C-49) package. The output from MOCUS is compatible with KITT (p. C-49) input requirements for quantitative reliability evaluation.

References

J. B. Fussell, E. B. Henry, and N. H. Marshall, Minimal Sets from Fault Tree, ANCR-1156, August 1974.

W. E. Vesely and R. E. Narum, PREP and KITT: Computer Codes for the Automatic Evaluation of a Fault Tree, IN-1349, August 1970.

J. B. Fussell, Synthetic Tree Model--A Formal Methodology for Fault Tree Construction, ANCR-1098, March 1973.

PEPSE

Purpose

PEPSE is a performance evaluation code that performs integrated energy balances for entire systems to determine power system efficiencies or for design studies.

Status

A.	Development	Developed by Energy Incorporated, Idaho Falls, Idaho.
B.	Availability	Proprietary.
C.	Utilization	Has been used by utilities to evaluate plant performance and assess planned modifications to increase overall efficiency.

Problems Solved

1. Performs steady-state heat balances on both the heat source and turbine generator cycles, either separately or combined, to determine total plant performance.
2. Parametric studies of plant equipment can be performed.

Systems and Subsystems Simulated

Total system performance.

Modeling

1. A network of components and streams describes the physical geometry of the plant.
2. The plant model consists of components that describe the hardware items such as tees, turbine stage groups, condensers, etc., and streams that describe the piping length or interconnection between components.
3. Can be used in either a performance or design mode or both.
4. The code output includes the thermal analysis of various plant components and generator electrical output.
5. Test data, whenever possible, can be used instead of design information.

Computer Operations

1. Operable on the CDC 6600 in a batch mode. Written in FORTRAN IV.
2. Typical run time for a heat source and turbine cycle heat balance for a 1,000 MWe plant is about 20 to 35 seconds.

Reference

F. D. Lang and W. C. Kettenacke, *PEPSE™: A Method for Power Plant Performance Evaluation* (Idaho Falls, Idaho: Energy Incorporated, September 1979).

Contact

Energy Incorporated
P.O. Box 736
Idaho Falls, Idaho 83401

PLENUM

Purpose

PLENUM calculates the bulk flow distribution in a cylindrical reactor coolant inlet plenum during a steady-state condition.

Status

A.	Development	Developed by E.I. Du Pont de Nemours at the Savannah River Laboratory, Aiken, South Carolina.
B.	Availability	Released to the National Energy Software Center (NESC) in 1972. Cost $61.00.
C.	Utilization	Used to evaluate flow distribution in lower or inlet plenums for input into core thermal-hydraulic and emergency core cooling (ECC) penetration calculations.

Problems Solved

Calculates the bulk flow distribution, velocity distribution, and other hydrodynamic conditions in the reactor inlet plenum.

Modeling

1. The fluid flow is calculated at specified mesh points by potential flow theory.
2. Written as two separate programs:
    a. Part 1 -- The Laplace equation for the velocity potential is solved for general, symmetric boundary conditions and one inlet nozzle by Gauss-Seidel relaxation.
    b. Part 2 -- Nonsymmetric, dimensional solutions are formed by superposition of the general solution (Part 1).
3. Three-dimensional effects are calculated so that asymmetric fluid input and output problems may be solved.
4. Use of a linear field equation permits rapid solution by superposition of many combinations of input and output boundary conditions.
5. First-order effects of friction may be included by addition of constant coefficients in the finite difference formulation.

Limitations

1. Dimensioned for a maximum of 8,000 mesh points in an (R, Theta, Z) array. Dimensions can be expanded at the expense of run time and core size.

2. Details of flow, such as boundary layer effect, are neglected.

Computer Operations

1. Operable on the IBM 360 using OS/360. Written in FORTRAN IV.

2. Requires 240,000 bytes of storage as well as an on-line punch and printer.

3. Program run time for Part 1 is proportional to N**2, where N is the number of mesh points. With 2,299 mesh points, CPU time is about 42 seconds on the IBM 360/195. Run time for Part 2 is about 1 second per problem.

4. Part 1 provides a general dimensionless solution to the field equation. For a given header, this program need only be executed once. Part 2 uses punched output from Part 1 to determine solutions for a variety of user specified boundary conditions.

Reference

S. D. Harris and J. W. Reece, "Application of Potential Flow Theory to Reactor Coolant Circulation," Proceedings of the Eight Southeastern on Thermal Sciences, DP-MS-71-13, 23 to 24 March 1972.

PREP/KITT

Purpose

PREP/KITT is a computer package developed to obtain system reliability information from a fault tree.

Status

A.	Development	Developed by Aerojet Nuclear Corporation at the Idaho National Engineering Laboratory, under employment by EG&G Idaho, Inc.
B.	Availability	Released to the National Energy Software Center (NESC) in 1972. Cost $70.00 (IBM) and $66.00 (UNIVAC).
C.	Utilization	Used to obtain reliability information on nuclear power plant systems and components.

Problems Solved

1. PREP obtains minimum cut sets and minimum path sets for a system fault tree.
2. KITT obtains exact reliability quantities as a function of time rather than single point values for the system or components of each minimal cut or path set.

Modeling

1. PREP obtains minimal cut sets by either direct deterministic testing or by an efficient Monte Carlo algorithm. The minimal path sets are obtained using the Monte Carlo algorithm.
2. Reliability results from KITT are obtained by upper bound approximations or by a bracketing procedure in which various upper and lower bounds may be obtained as close to one another as desired. The reliability information is obtained from numerical solution of the simple integral balance equations of kinetic tree theory.
3. KITT can handle independent components which are nonrepairable or which have a constant repair time. Any assortment of nonrepairable components or components having constant repair times can be considered.
4. Any inhibit conditions having constant probabilities of occurrence can be handled.

5. The failure intensity of each component is assumed constant with respect to time. Components which, during different time intervals (called phases), have different reliability properties can be considered.

Limitations

1. PREP obtains minimum cut sets and/or path sets for any tree having a maximum of 2,000 components and 2,000 logical gates. All gates of the fault tree must be AND and OR gates, and no gate may have more than 7 inputs consisting of both components and other gates. PREP can obtain, at most, 1,000 minimal cut sets with a maximum of 19 components per set and, at most, 500 minimal path sets with a maximum of 29 components per set.

2. KITT will obtain reliability information for up to 500 minimal cut or path sets containing a total of, at most, 400 distinct components and inhibit conditions. Reliability information for each of 50 different phases per component can be obtained.

Computer Operations

1. Operable on the IBM 360 using OS/360 MUT and the UNIVAC 1108. Written in FORTRAN IV.

2. Requires 350,000 bytes of memory on the IBM 360.

3. Run time on the IBM 360/75 is about 0.06 second for a ten-component tree without repair.

Interface/Overlap

KITT 1 and KITT 2 are modifications of the basic KITT program.

References

W. E. Vesely, Analyses of Fault Trees by Kinetic Tree Theory, IN-1330, October 1969.

W. E. Vesely and R. E. Narum, The Users Manual for the PREP and KITT Codes, IN-1344, August 1970.

W. E. Vesely, "Single Precision Modifications to the KITT 1 and KITT 2 Fault Tree Codes," Idaho Nuclear Corporation Memorandum, WEV-54-70, December 1970.

PROSA

Purpose

PROSA was developed for use with accident analysis programs to conduct probability studies of hypothetical accidents.

Status

A.	Development	Developed by the Reactor Analysis and Safety Division at the Argonne National Laboratory, Argonne, Illinois. The current version is PROSA-1.
B.	Availability	Released to the National Energy Software Center (NESC) in 1978. Restricted to use by U.S. citizens for either U.S. Government or commercial nonmilitary applications at a U.S. installation. Cost $64.00.
C.	Utilization	Has been used in the safety analysis of LMFBRs but can be applied to LWR safety considerations.

Problems Solved

1. Determines the distribution of a random variable that is a function of many other random variables when this functionality is not known analytically and can be obtained only through parametric studies with a deterministic computer code.

2. PROSA-1 calculates probability histograms and joint histograms for the output variables.

Modeling

1. PROSA first systematically provides input parameter combinations for evaluation by a deterministic code. Then, with the output from the deterministic code for each of these parameter combinations, multivariate quadratic response surfaces are generated approximating the functionality between the input and output of the deterministic code.

2. The first part of PROSA determines knot-point coordinates of the parameters, using a user-specified probability range (confidence interval). These serve as input to deterministic codes, which calculate consequence values in the specified knot points.

3. The consequence values are used by the second part of PROSA-1 to solve

   a. The approximating functions (response surfaces),

   b. The sensitivity/importance of each parameter with respect to the consequence variables,
   c. The statistical moments of the input parameters,
   d. The mean values and standard deviations of the consequences, and
   e. The correlation coefficients for all pairs of the consequences. In the simplest case, a single multivariate second-degree surface is matched to the knot-point consequences.

4. By use of random-number sampling of the parameter distributions and simulation with the as-calculated response surface, the program also calculates

   a. The probability distributions and the first four moments of the consequences,
   b. The joint distributions, and
   c. The statistical-error estimates for the distribution.

5. It is possible to use regionwise response surfaces, distinct second-degree surfaces for every "quadrant" of the multivariate parameter space.

6. By defining one or more input parameters to be consequences as well, it is possible to use the code for calculating optimal importance distributions for the input parameters.

7. As input data, PROSA needs the probability distributions of the variable parameters (e.g., reactivity and heat transfer coefficients considered as sources of uncertainties in an accident analysis). Uniform, normal, truncated normal, exponential, beta, and log-normal probability distributions are available.

8. Special features of the code generate or treat sensitivities, statistical moments of the input and output variables, regionwise response surfaces, correlated input parameters, and conditional distributions.

Limitations

1. The maximum number of variable input parameters and consequence variables that can be analyzed simultaneously is 12 and 6, respectively.

2. Eight different probability distributions are available for input parameters.

3. The correlations (if any) between the input parameters are limited to linear correlations.

4. A maximum of four simultaneous criteria or conditions can be specified for the consequences.

5. The individual and joint distributions are obtained in forms of histograms with 12 and 144 categories, respectively. The width of the categories is a user-supplied fraction of the standard deviation of the consequences.

Computer Operations

1. Operable on the IBM 370 using OS/370. Written in FORTRAN IV (97%) and Assembly language (3%).
2. Requires 350,000 bytes of storage and system input/output and punch units.
3. The typical computer run time for a case of six input parameters and six consequence variables is about 60 seconds for 40,000 simulations on the IBM 370/195.
4. Contains a few FORMAT statements using T format (an IBM extension of ANSI FORTRAN).

Interface/Overlap

PROSA-1 is designed for use in conjunction with a deterministic/mechanistic accident analysis code, such as RELAP (p. A-95). The first part of PROSA-1 provides input to the accident analysis code, which, in turn, provides input to the second part of PROSA-1.

References

J. K. Vaurio and C. Mueller, "Probabilistic Analysis of Liquid-Metal Fast Breeder Reactor Accident Consequences with Response Surface Techniques," Nuclear Science and Engineering, Volume 65 (1978), pp. 401-413.

J. K. Vaurio and C. Mueller, PROSA-1: A Probabilistic Response Surface Analysis Code, ANL/RAS 77-35, September 1977.

RAS

Purpose

RAS is a reliability analysis system composed of an integrated package of computer programs for the quantification of fault trees.

Status

A.	Development	Developed by EG&G Idaho, Inc. at the Idaho National Engineering Laboratory, Idaho Falls, Idaho.
B	Availability	Operational. Scheduled for release to the National Energy Software Center (NESC).
C.	Utilization	Used in reliability and safety analysis and as an aid in the analysis of phased missions.

Problems Solved

The RAS package of codes has been integrated to find minimal cut sets from fault trees, calculate reliabilities, quantitatively reduce the number of minimal cut sets, and perform common cause and common mode failure analysis.

Modeling

1. The RAS package consists of MOCUS, POCUS, KITT-1, SRTPRN, COMCAN, and FATRAM. A brief description of each of these codes follows.

    a. MOCUS

    MOCUS is used to determine the minimal cut sets from a fault tree. Only AND and OR gates are allowed. The algorithm used to obtain the cut sets begins with the top event of the fault tree. The top event is replaced by the gates and the basic events that are input to it. If the top event is an AND gate, the inputs are listed in a row. If the top event is an OR gate, the inputs are listed in a column. Each gate in the matrix is now expanded using the same substitution technique. The process continues until only basic events are contained in the matrix. The rows of this final matrix are the cut sets of the tree. Minimal cut sets are obtained by eliminating supersets from the matrix. The order of a cut set is the number of events in the cut set. Output from MOCUS consists of an input edit, plus all of the cut sets up to a desired order. In addition to obtaining cut sets for the top event, cut sets can also be obtained for intermediate events in the tree.

b. **POCUS**

POCUS is a computer program used to calculate reliability characteristics for the basic events, for the cut sets, and for the top event of a fault tree. The input to POCUS consists of the failure rates and repair rates of the basic events and the minimal cut sets for the system. Output from POCUS consists of bounds on the unavailability, unreliability, or expected number of failures, and the failure rate for the basic events, for the cut sets, and for the top event.

c. **KITT-1**

KITT-1 is a computer program used to calculate reliability characteristics for the basic events of the cut sets and for the top event of a fault tree. Input to KITT-1 is similar to that of POCUS. KITT-1 will calculate the "min cut" upper bound and will also perform "bracketing" on the reliability characteristics. Bracketing yields bounds that are successively closer to the exact results.

d. **SRTPRN**

SRTPRN is a computer program that sorts minimal cut sets into groups according to the importance of the cut sets. Three criteria are used to determine the importance of a cut set:

(1) Shorter cut sets are usually more important than those with a larger number of basic events.

(2) Among cut sets of a particular size, cut sets containing more active basic events (with higher failure probabilities) are more probable than those with more passive events (events with low failure probabilities).

(3) Cut sets composed of the same type of basic events or components are more important than cut sets containing different types of components, due to the increased potential for common mode failures.

Cut sets are printed according to the above criteria. If desired, unimportant cut-set groups are discarded or "pruned."

e. **COMCAN**

COMCAN is a computer program used in common cause failure analysis. COMCAN searches basic events in minimal cut sets for shared secondary failure event susceptibilities or common linking conditions among components. Secondary events have been classified in three generic common cause categories. If the events in a cut set share a generic common-cause susceptibility, the cut set can then be searched to see if all the basic events also share a common location.

f. FATRAM

The FATRAM algorithm is used for the determination of minimal cut sets. Basically, it is a downward algorithm like MOCUS, but the following rules are used to determine the order in which gates are processed:

(1) AND gates are resolved first,
(2) OR gates with other inputs are resolved second, and
(3) OR gates with only basic event inputs are processed last.

Routines to handle repeated events and also other special computing techniques are used to minimize the amount of core needed for processing a tree.

2. Although the code package was designed to analyze phased missions, single-phase fault trees can also be analyzed. These features allow the user to calculate time-dependent bounds on mission unreliability.

Limitations

1. A phased mission with a maximum of five distinct phases can be analyzed. Each phase is normally analyzed in a sequential manner.
2. Path sets can be obtained from MOCUS only, but no reliability characteristics can be calculated using the path set.

Computer Operations

1. Operable on the CDC 7600 computer. Written in FORTRAN.
2. A minimal cut-set problem, utilizing the FATRAM routine with 28 gates and 19 basic events with a maximum cut-set order requested of 10, required 22,000 words of memory and about 1 second of computer time.

References

D. M. Rasmuson, N. H. Marshall, and G. R. Burdick, User's Guide for the Reliability Analysis System (RAS), TREE-1168, September 1977.

J. B. Fussell, E. B. Henry, and N. H. Marshall, MOCUS: A Computer Program to Obtain Minimal Sets from Fault Trees, ANCR-1156, August 1974.

J. B. Fussell et al., SUPERPOCUS: A Computer Program for Calculating System Probabilistic Reliability and Safety Characteristics, (Tennessee: University of Tennessee, May 1977).

W. E. Vesely and R. E. Narum, *PREP and KITT: Computer Codes for the Automatic Evaluation of a Fault Tree*, IN-1349, August 1970.

C. A. Dolan, *SRTPRN: A Program to Sort and Prune Fault Tree Cut Sets*, TREE-1012, December 1976.

G. R. Burdick, N. H. Marshall, and J. R. Wilson, *COMCAN: A Computer Program for Common Cause Analysis*, ANCR-1314, June 1976.

RELO1

Purpose

RELO1 computes the failure probability for a single failure mode.

Status

A.	Development	Developed by the Westinghouse Electric Corporation at the Bettis Atomic Power Laboratory, West Mifflin, Pennsylvania.
B.	Availability	Released to the National Energy Software Center (NESC) in 1971. Restricted to use by U.S. citizens for either U.S. Government or commercial nonmilitary application at a U.S. installation. Cost $57.00.
C.	Utilization	Used in determining failure probabilities to assist fault tree analysis of reactor safety considerations.

Problems Solved

Calculates the density of the failure probability distribution and the average failure probability.

Modeling

1. The failure probability is calculated by means of a simplified Monte Carlo technique that randomly selects a value for the applied stress and calculates the probability that the strength is smaller than the stress.

2. Two options are available:

   a. Calculation of the interaction of two normally distributed variates, and
   b. Calculation of the interaction of two truncated, normally distributed variates.

3. Calculates the interference of two truncated normal distribution functions.

Limitations

The number of random selections of the applied stress is restricted to 2,000.

Computer Operations

1. Operable on the CDC 6600 using Scope 31. Written in FORTRAN IV.
2. Requires 110,000 (octal) words of memory and standard input, output units.
3. Run time is less than 1 minute.
4. Uses the Bettis Environmental Routine.

References

D. R. Rauth, C. M. Smith, and E. C. Strothers, RELO1 - A Reliability Program for a Single Failure Mode, WAPD-TM-1009, May 1971.

C. J. Pfeifer, CDC-6600 FORTRAN Programming - Bettis Environmental Report, WAPD-TM-668, January 1967.

J. D. Church and B. Harris, "The Estimation of Reliability From Stress-Strength Relationships," Technometrics, 12 (1970), pp. 49-54.

Z. Govindarajulu, "Distribution-Free Confidence Bounds For P(X less than Y)," Ann. Inst. Stat. Math., 20 (1968), pp. 229-238.

D. B. Owen, K. J. Craswell, and D. L. Hanson, "Nonparametric Upper Confidence Bounds For PR(Y less than X) and Confidence Limits for P(Y less than X) when X and Y are Normal," Am. Stat. Assoc. J., 59 (1964), pp., 906-924.

REPP

Purpose

REPP provides a method for evaluating core and fuel rod performance and design during steady-state conditions.

Status

A.	Development	Developed by Battelle Northwest Laboratory, Richland, Washington.
B.	Availability	Released to the National Energy Software Center (NESC) in 1970. Cost $64.00.
C.	Utilization	Used for nuclear power plant total core and fuel rod design and evaluation of design margins in heat transfer, fluid flow, and geometry.

Problems Solved

1. Evaluates fuel temperatures and critical heat flux margins for a fixed reactor core and fuel design.
2. Determines the number of fuel pins required to maintain specified heat flux margins from burn-out at a given reactor power level.
3. Determines the fuel pin diameter to design within fuel centerline temperature limits at a specified reactor power level.
4. Evaluates the centering effect on fuel temperatures.
5. Evaluates the code pressure drops and coolant properties for single-phase and two-phase flow for fuel operating at average reactor conditions and at a theoretical hot-pin, hot-channel condition.
6. Calculates pressure drops across several types of fuel pin spacers.

Systems and Subsystems Simulated

Nuclear reactor core and fuel elements.

Modeling

1. Represents the core as a right-circular cylinder containing fuel pins arrayed uniformly in a triangular or square lattice. The model is segmented into axial nodes and radial fuel pin nodes.

2. A flux depression may be used in the input which is independent of the radial nodal mesh utilized.

3. The effects of fuel centering on fuel temperatures is optional and, when used, the void diameter at the center of the fuel is determined.

4. Nominal and hot channel thermal and hydraulic conditions are predicted at each nodal location based on an energy and momentum balance.

5. Steam tables are provided along with a variable fuel thermal conductivity insert. The local energy generated and transferred is determined from input consisting of the reactor power and a normalized axial power profile. The axial power distribution specified may be independent of the nodal mesh utilized. A Lagrangian interpolation routine is used to expand the input array to be consistent with the nodal mesh.

6. Integration is performed when required using Simpson's Rule.

7. The fuel spacer model uses the A. N. Destardeur technique to predict the pressure drop across wire wrap, honeycomb, lenticular wire, or wire-type spacers, at the option of the user.

Limitations

1. A minimum of 10 and a maximum of 100 axial mesh nodes. A maximum of 21 radial fuel pin nodes.

2. The effect of the restricted coolant flow areas in the corners of encased fuel assemblies (an inherent flow problem especially in two-phase flow) is not calculated.

3. The cold wall influence on burn-out heat fluxes is not evaluated.

4. Interchannel mixing and flow distribution present some uncertainty; however, since the sizing technique is theoretically based, these restrictions need not be serious, providing sufficient data on fuel performance is known to provide input to the code.

5. Limited at present to water-cooled reactors.

6. The W-2 and W-3 burn-out correlations presently used can be replaced to accommodate other burn-out correlations.

Computer Operations

1. Operable on the UNIVAC 1108 using CSCX. Written in FORTRAN IV.

2. Requires 51,000 bytes of addressable core storage. There are 13 routines in REPP and no overlay links. Input is by card batch.

3. Run time is dependent upon problem complexity. If initial conditions are relatively close to the solution, the run time may be as short as 8 to 10 seconds. Otherwise, the iterative solution may require about 30 seconds. A single pass through the code requires 2 to 3 seconds.

Reference

R. M. Hiatt and C. Bromley, Jr., REPP - A Thermal Hydraulic Design Code for Water-Cooled Reactors, BNWL-1013, March 1969.

SAFTAC

Purpose

SAFTAC is a Monte Carlo fault tree simulation program that provides a systematic approach for analyzing systems.

Status

A. Development	Developed by United Nuclear Industries, Inc., Richland, Washington.
B. Availability	Released to the National Energy Software Center (NESC) in 1979. Cost $61.00.
C. Utilization	Used in design and safety analysis of reliability and consistency for complex systems.

Problems Solved

1. The fault tree technique provides analysis flexibility that ranges from equipment analysis to overall plant analyses, incorporating all of the influencing elements on a total-system basis.

2. The deductive approach used in this technique is useful for evaluating design consistency and reliability, for judging alternatives, for determining acceptability of trade-offs, and for analyzing multiple failure combinations and common mode failures in complex systems.

Modeling

1. Assumes an exponential failure distribution for basic input events and a choice of either Gaussian-distributed or constant repair times.

2. Views the system represented by the fault tree as a statistical assembly of independent basic input events, each characterized by an exponential failure distribution and, if used, a constant or normal repair distribution.

3. The fault tree method allows analyses to consider the rate at which failures or events are detected after they occur (detection time) and the rates at which they are restored to normal (repair time). The fault tree method is concerned with assuring that all critical activities are identified and eliminated or controlled.

4. A system, when defined in terms of the all-encompassing analysis capability of fault tree analysis, is a composite of operational and support equipment, personnel, facilities, and software, which

are used together as an entity and are capable of performing and/or supporting an operational role.

Limitations

1. Allows a maximum of 1,100 basic events and 1,100 logical gates.

2. Can be redimensioned to handle 2,000 basic events and 2,000 logical gates within the existing core memory requirement.

Computer Operations

1. Operable on the UNIVAC 1108 using the CSCX or EXEC 2 operating system. Written in FORTRAN IV and SLEUTH II. (An IBM version is also available.)

2. Run time is dependent upon the fault tree size and structure and the degree of resolution desired in the quantitative evaluation.

References

Paul A. Crosetti, Fault Tree Simulation Computer Program, DUN-7697, June 1971.

P. A. Crosetti and R. A. Bruce, "Commercial Application of Fault Tree Analysis," Ninth Reliability and Maintainability Conference, Annals of Reliability and Maintainability, Volume 9 (1970), p. 230.

P. A. Crosetti, Fault Tree Analysis with Probability Evaluation, DUN-SA-157, 3 November 1970.

P. A. Crosetti, Fault Tree Analysis for Reactor Systems, DUN-SA-165, 25 February 1971.

P. A. Crosetti, Computer Program for Fault Tree Analysis, DUN-5508, April 1969.

Excerpt from CSCX RJE/CRJE Reference, Volume 2: Library Elements, E00003-02-00, April 1971.

SETS

Purpose

SETS provides for the evaluation of fault trees for nuclear safety systems.

Status

A.	Development	Developed at Sandia National Laboratories, Albuquerque, New Mexico, for the Department of Energy.
B.	Availability	Released to the National Energy Software Center in 1978. Cost $66.00.
C.	Utilization	Used to evaluate fault trees of nuclear safety systems.

Problems Solved

Determines minimum cut sets, i.e., all the fundamental ways that an intermediate event can occur, and also determines cut sets for top events. Can be used for common cause analysis.

Modeling

1. Uses the Set Equation Transformation System for fault tree analysis.
2. The occurrence of the top event or any intermediate event of a fault tree is represented by a Boolean equation.
3. This Boolean equation is evaluated to determine the minimal cut sets of the event.
4. For large fault trees, the independent subtrees and stem equations are determined separately. These results are then combined to evaluate the large tree (processing of fault trees in stages).
5. SETS uses a variety of predetermined gates, including the standard AND and OR gates as well as the INHIBIT, PRIORITY AND, and EXCLUSIVE OR gates. User-supplied special gates for any valid logic function can be used. These can contain an event and its complement.
6. Includes a truncation option of the cut-sets equation that cannot be derived exactly and produces the most important minimal cut sets.

7. Several methods for fault tree evaluation are incorporated, including a method for common cause analysis.

Limitations

1. The number of gates and events in a fault tree is not limited by the program but rather, by the computer memory.
2. For trees larger than 50 gates, the recommended analysis is to evaluate independent subtrees and stem equations and then combine for the larger trees.

Computer Operations

1. Operable on the CDC 6600 using Scope 3.3 and can be used on the CDC 7600 using Scope 2.1. Written in FORTRAN extended language.
2. Requires 205,000 (octal) words of core storage.
3. Typical run time using the subtree and stage methods for a tree with 44 events and 43 gates producing 1,053 minimal cut sets is about 30 seconds.

References

R. B. Worrell and D. W. Stack, *A SETS User's Manual for Fault Tree Analysis*, NUREG/CR-0465, SAND77-2051 (Albuquerque: Sandia Laboratories, November 1978).

R. B. Worrell, *Set Equation Transformation System (SETS)*, SLA-73-0028A (Albuquerque: Sandia Laboratories, May 1974).

R. B. Worrell, *Using the Set Equation Transformation System in Fault Tree Analysis*, SAND74-0240 (Albuquerque: Sandia Laboratories, September 1974). Also condensed in "Reliability and Fault Tree Analysis," *SIAM*, September 1975.

R. B. Worrell and D. W. Stack, *Common-Cause Analysis Using SETS*, SAND77-1832 (Albuquerque: Sandia Laboratories, December 1977).

TAFY

Purpose

TAFY analyzes the conditions within a fuel rod during the lifetime of the fuel.

Status

A.	Development	Developed by Babcock and Wilcox (B&W), Nuclear Power Generation Division, Lynchburg, Virginia.
B.	Availability	Proprietary.
C.	Utilization	Has been used to analyze the lifetime behavior of UO_2 oxide fuel pellets and cladding for the safety analysis report (SAR) analysis of B&W plants built for the Tennessee Valley Authority (TVA). Code calculations have been compared with data from the HALDEN reactor experimentation in Norway.

Problems Solved

Determines the following:

1. The radial temperature distribution of the fuel at each axial location,
2. Coolant temperature,
3. Average and maximum temperatures,
4. Fuel-cladding gap conductance and gas conductivity,
5. Fuel-cladding contact pressure,
6. Fuel swelling at each location,
7. Fuel and cladding dimensions,
8. Fraction of the hot fuel diameter that is contained in the gain growth regions, molten fuel, and central void,
9. Densification,
10. Internal pin pressure,
11. Number of fission gas atoms yielded, and
12. Fission gas plenum volume.

Systems and Subsystems Simulated

A nuclear fuel rod and internals.

Modeling

1. The cylindrical fuel rod is divided into radial and axial increments.
2. Fuel rod conduction accounts for temperature-dependent properties, fuel dimensional changes, and centerline melting.
3. The fuel-cladding head transfer considers pressure and contact effects.
4. Fuel and cladding dimensional changes consider thermal expansion gain growth characteristics, internal and external pressure differences, and fission product accumulation.
5. The constitutive equations are in a finite difference form and solved by an iterative technique.
6. Linear heat rate and dimensional information must be input.

Limitations

1. The maximum number of axial increments allowed is 70.
2. The number of radial increments may vary from 10 to 30.
3. The radial temperature distribution is independent of the number of radial increments if the fuel density is radially uniform.
4. No axial conduction is considered.
5. The code is a steady-state code; no transient behavior is modeled.
6. The accuracy of the gas pressure calculation is improved with more radial increments.
7. The maximum number of lifetime steps is 40.

Computer Operations

Operable on the CDC 6600. Written in FORTRAN.

References

C. D. Morgan and H. S. Kao, TAFY - Fuel Pin Temperature and Gas Pressure Analysis, BAW-10044, April 1972.

G. Kjaerheim and E. Rolstadt, IN-Pile Determination of UO_2 Thermal Conductivity, Density Effects and Gap Conductance, HPR-80, OECD, Halden Reactor Project, 1967.

[redacted]

THI3D

Purpose

THI3D performs a steady-state thermal-hydraulic analysis of a multichannel fuel assembly or a multiassembly core.

Status

A.	Development	Developed by Argonne National Laboratory (ANL), Components Technology Division, Argonne, Illinois.
B.	Availability	Released to the National Energy Software Center (NESC) in 1976. Cost $72.00.
C.	Utilization	Used for design studies and analysis of heat transfer and fluid flow in nuclear cores.

Problems Solved

Determines temperature distributions in the coolant, cladding, fuel, and duct wall and the size of the central void in the oxide fuel after thermal reconstruction.

Systems and Subsystems Simulated

Multichannel and pin fuel assembly or multiassembly core of a nuclear reactor.

Modeling

1. Each fuel channel or assembly is modeled individually and segmented with unlimited axial steps.
2. Three different thermal-hydraulic channel arrangements (square, hexagonal, and triangular) can be analyzed.
3. The model accounts for conservation of mass, energy, and momentum (both axial and transverse), subject to pressure-drop boundary conditions.
4. Multipin analysis with transverse interactions or multiassembly calculations without transverse interactions between the channels can be performed.
5. Turbulent interchange, radial thermal conduction, and forced flow due to wire wrap or grids between channels are explicitly taken into account. A core fluid bypass option is available.

6. Problems of partial flow blockage (porous blockage) can be analyzed, and the effect on the redistribution of the inlet mass velocity can be taken into account.

7. Solves the multipoint boundary value rod-bundle thermal-hydraulic problems by means of a mixed implicit and explicit finite differencing of conservation of mass, energy and momentum equations. The Newton-Raphson method is employed for iteration on the boundary conditions.

Limitations

1. A 217-pin bundle or assembly is the maximum allowed.

2. The fluid is single phase with no treatment of two-phase conditions.

3. Sonic-velocity propagation (water-hammer) effects are not considered.

4. A lumped-parameter approach is assumed where a reactor channel may be either a fuel pin or a fuel assembly with associated coolant.

5. Considers only steady-state behavior; therefore, transient situations cannot be considered.

6. All system parameters within a channel, at a given axial location, are considered uniform and radially isentropic. Axial heat conduction is neglected.

Computer Operations

1. The code is operable on the IBM 370 using OS/MUT. Written in FORTRAN IV.

2. Requires 260,000 bytes of storage (for a 19-pin bundle) and two scratch tapes or disks. Large problems (61 pins and 217 pins) may be analyzed by changing the common blocks in the block data subprogram.

3. A 19-pin rod-bundle problem required about 1.5 minutes of execution time on the IBM 370/195.

4. The ANL version uses the following three routines, which may be deleted or replaced by similar routines used at other installations:

   a. ERRSET Allows user processing upon underflow, overflow or divide check; used for debugging,
   b. TLEFT A time function that returns the amount of time left in a run; used for timing and restart control, and

c. LOCF A location function that returns the absolute address of the argument; used for clearing large blocks of storage.

References

W. T. Sha and R. C. Schmitt, THI3D--A Computer Program for Steady-State Thermal-Hydraulic Multichannel Analysis, ANL-8112, December 1975.

W. T. Sha, R. C. Schmitt, and T. R. Huebotter, "Boundary-Value Thermal-Hydraulic Analysis of a Reactor Fuel Rod Bundle," Nuclear Science and Engineering, Volume 59 (1976), pp. 140-160.

TORC

Purpose

TORC determines the thermal margin (minimum departure from nucleate boiling ratio) for steady-state operation of a open-lattice PWR core.

Status

A.	Development	Developed by Combustion Engineering Incorporated, Windsor, Connecticut.
B.	Availability	Proprietary.
C.	Utilization	Used for design studies of fuel rods, fuel assemblies, and core. Results have been compared to data from the Columbia CHF tests.

Problems Solved

Determines flow channel exit enthalpy, temperature, density quality, void fraction, flow, and mass flux. Determines the following quantities versus axial distance:

1. Pressure drop.
2. Enthalpy.
3. Flow rate and mass flux.
4. Local to average mass flux ratio.
5. Heat flux.
6. Channel-to-channel diversion cross-flow.
7. DNBR (departure from nucleate boiling ratio) for each fuel rod if desired.

Systems and Subsystems Simulated

Nuclear core, fuel assemblies and fuel rods with associated flow channels.

Modeling

1. A three-dimensional representation of an open-lattice core is used to determine the local coolant conditions at all points in the core. The calculation is performed in three steps:

  a. Calculation of the coolant conditions throughout the core on a coarse mesh basis. A symmetrical core quadrant is normally modeled, and the smallest unit represented by a flow channel is a fuel assembly.
  b. The local coolant conditions are determined in the limiting hot subchannel in two stages. In stage 1, the hot assembly is analyzed with a coarse mesh, in which the hot fuel assembly and the immediate neighboring fuel assemblies are modeled. In stage 2, the partial assembly region containing the hot subchannel is modeled with a fine mesh where the hot subchannel, a large number of surrounding subchannels, and the fuel rods are individually simulated.
  c. DNBR is calculated for each subchannel/flow channel using the local coolant conditions.

2. Assumes steady-state, one-dimensional, single-phase, or homogeneous two-phase flow.

3. Solves the conservation of mass, energy, and momentum equations accounting for the conservation of lateral momentum and nonzero lateral boundary conditions.

4. Accounts for specific phenomena, using the following models and correlations:

  a. Forced convection heat transfer uses the Dittus-Boelter correlation and nucleate boiling heat transfer uses the Jens-Lottes correlation.
  b. Axial pressure drop accounts for cross-flow momentum and turbulent interchange. Two-phase pressure drop uses either the homogeneous model, the Armand correlation, or the Sher-Green and Modified Martinelli-Nelson correlations.
  c. Flow resistance due to spacer grids and cross-flow is accounted for.
  d. Critical heat flux is calculated using either the B&W 2 CHF correlation, the updated W-3 CHF correlation, the original W-3 CHF correlation, or the CE-1 CHF correlation.

5. Requires geometry and heat generation of the fuel assemblies as input to the model as well as the core inlet flow and radial temperature distribution.

Limitations

1. TORC is steady-state and does not calculate transient conditions.

2. The maximum number of flow channels is limited to 55.

3. The maximum number of flow channel connections is 99. Each flow channel can have several connections with other flow channels.

Computer Operations

Operable on the CDC 6600. Written in FORTRAN.

References

TORC CODE - A Computer Code for Determining the Thermal Margin of a Reactor Core, CENPD-161, Combustion Engineering, Incorporated, July 1975.

O. J. Mendler, A. S. Rathbun, N. E. Van Huff, and A. Weiss, "Natural Circulation Tests with Water at 800 to 1200 psia Under Non-boiling Local Boiling and Bulk Boiling," *Journal of Heat Transfer* (August 1961), pp. 261-273.

T. M. Burke, C. E. Meyer, and J. Shefcheck, *Analysis of Data from Zion (Unit 1) THINC Verification Tests*, WCAP-8454, December 1974.

WAMCUTS

Purpose

WAMCUTS produces minimum cut sets for a given fault tree.

Status

A.	Development	Developed by Science Applications Inc., for the Electric Power Research Institute, Palo Alto, California.
B.	Availability	Proprietary.
C.	Utilization	Used to evaluate fault tree minimum cut sets.

Problems Solved

1. Evaluates fault trees by obtaining the minimal cut sets and computing the unavailability of the events.
2. Can be used to determine the mean value of the top-of-the-tree event probability as well as the probability distribution and information on the most likely event paths.

Modeling

1. Checks for input error, illegal gates, and syntax.
2. Three types of gates (AND, OR, and NOT) and combinations of these gates are considered allowing any complex system to be modeled.
3. Restructures the tree into gates with no more than two inputs and forms new gates while adjusting original gates.
4. A user option for NOT gates allows tree pruning by setting the first moment equal to unity.
5. To form the cut sets of a gate from the cut set of its input, the operation of ANDing, ORing and NOTing of the cut sets of two inputs is performed. These operations are a functionally complete set of logic operations, i.e., any of the 16 Boolean functions of 2 variables can be formed from these 3 operations.
6. The option of eliminating cut sets with probability of occurrence below an input value is available.

Limitations

1. Limited to trees with not more than 1,500 gates and 1,500 components.

2. There is no limitation on the size of the cut sets, although the number of cut sets per gate is limited to 2,000.

Computer Operations

1. Operable on the CDC 7600 using Scope. Written in extended FORTRAN IV.

2. Requires the FTN4 compiler and written for the CDC 60 bit word.

3. Run time is from 4 to 125 seconds, depending upon the program size and the output wanted. A 298-gate, 124-component problem requires 4.82 seconds to determine cut sets, 47.09 seconds to determine first and second moments, and 2.1 seconds for the percent error of the first moment. The user-specified minimum probability for this problem was 10^7.

Interface/Overlap

WAMCUTS is part of the WAM family of computer codes.

References

R. C. Erdman et al., WAMCUTS, A Computer Code for Fault Tree Evaluation, NP-803, Project 767-1 (Palo Alto: Electric Power Research Institute, June 1978).

F. L. Leverenz, Jr. and H. R. Kirch, User's Guide for the WAMBAM Computer Code, 277-2-5 (Palo Alto: Electric Power Research Institute, January 1976).

APPENDIX D

Basic Engineering Program

The computer codes listed below perform general engineering calculations or provide data (steam and water properties) to other programs. The general engineering codes consist of one- , two- , and three-dimensional heat transfer and fluid flow programs including two-phase flow, equation solvers including ordinary and partial differential equations, and mesh generators along with other general engineering calculations. All of the computer codes listed have been released to the National Energy Software Center (NESC).

Computer Program	Functional Calculations
ASTM	Thermodynamic properties of steam and water
DE/STEP, INTRP	Solution of ordinary differential equations
DPOLE	Helmholtz equation in three dimensions
EISPACK2	Matrix eigenvalue/vector package
EPISODE	Solution of ordinary differential equations
EPISODED	Solution of ordinary differential equations with banded Jacobian
ERF, ERFC	Error and complimentary error function calculations
EVITS	Steady-state two-dimensional fluid flow calculations
FLAC	Steady-state flow and pressure distribution calculations
FLOW-MODEL	Multichannel, two-dimensional, two-phase flow calculations
FLOW3	Network analysis of three-dimensional, two-phase flow
FORSIM	Solution of partial and ordinary differential equations
GEAR, GEARB	Solution of ordinary differential equations with banded Jacobian
GNAT, MESH2, GPRINT	Two-dimensional nonlinear analysis

Computer Program	Functional Calculations
HEATING2	Transient and steady-state heat transfer analysis
HEATING5	One-, two-, and three-dimensional heat conduction program
HOT2	Two-dimensional transient heat conduction program
K-FIX	Transient two-dimensional, two-phase flow analysis
LIZARD	Solution of nonlinear differential equations
MO899, HOH	Steam tables, 14.5 to 2,538 psia
PDEONE	Solution of partial differential equations
QMESH, RENUM	Self-organizing mesh generator
SNEQ	Solution of nonlinear algebraic equations with curve plotter
SOLA-ICE	Transient compressible fluid-flow calculations
STEAM67	1967 ASME steam and water properties
SUPORT	Solution of boundary value problems
TAC2D, 3D	Steady-state and transient two- and three-dimensional heat transfer and temperature calculations
TRUMP	Calculation of transient and steady-state temperature distributions
WASP	Water properties for safety analysis
WATER	Steam tables, 14.7 to 14,000 psia and 32° to 472°F

APPENDIX E

Acronyms

This appendix contains a list of the acronyms commonly in use in the nucear reactor industry. Along with the acronyms used to label reactor systems, components and transient and other reactor and engineering related labels are included acronyms for national laboratories and research facilities, reactor code development companies, and foreign and U.S. Government entities.

Acronym	Definition
AAEC	Australia Atomic Energy Agency
AEC	Atomic Energy Commission
AEE	Atomic Energy Establishment
AI	Atomics International
ANC	Aerojet Nuclear Corporation
ANL	Argonne National Laboratory
ATWS	anticipated transients without scram
BAPL	Bettis Atomic Power Laboratory
BE	best-estimate model
BCL	Battelle-Columbus Laboratories
BNL	Brookhaven National Laboratory
BNWL	Battelle Northwest Laboratory
B&W	Babcock and Wilcox
BWR	boiling water reactor
CCFA	common cause fault analysis
CDA	core disruptive accident
CDC	Control Data Corporation
CE	Combustion Engineering, Incorporated
CEA	control element assembly
CFR	Code of Federal Regulations
CPU	central processing unit

Acronym	Definition
CS	containment system
DAEA	Danish Atomic Energy Agency
DNBR	departure from nucleate boiling ratio
DOE	Department of Energy
DPC	Duke Power Company
EBR-II	Experimental Breeder Reactor II
ECC	emergency core cooling
ECCS	emergency core coolant system
EG&G	Edgerton, Germeshausen and Grier, Inc.
ERDA	Energy Research and Development Agency
EI	Energy Incorporated
EM	evaluation model
EN	Exxon Nuclear, Inc.
EOS	Equation of state
EPRI	Electric Power Research Institute
FBR	fast breeder reactor
FFTF	Fast Flux Test Facility
FLECHT	full length emergency coolant heat transfer
FSAR	final safety analysis report
GA	General Atomic Corporation
GE	General Electric Company
GIT	Georgia Institute of Technology
GRS	Gesellschaft fur Reaktorsicherheit (F. R. Germany)
HEDL	Hanford Engineering Development Laobratory
IBM	International Business Machines, Inc.
INEL	Idaho National Engineering Laboratory
INC	Idaho Nuclear Corporation
JAERI	Japan Atomic Energy Research Institute
JNRC	Joint Nuclear Research Centre
KAPL	Knolls Atomic Power Laboratory
KE	Kaiser Engineers
KI	Kaiser Industries Corporation
LANL	Los Alamos National Laboratory
LBL	Lawrence Berkeley Laboratory
LLL	Lawrence Livermore Laboratory

Acronym	Definition
LMFBR	liquid metal fast breeder reactor
LOCA	loss-of-coolant accident
LOCE	loss-of-coolant experiment
LOF	loss-of-flow accident
LOFT	Loss-of-Fluid Test facility
LWR	light water reactor
MIT	Massachusetts Institute of Technology
MWe	megawatt electric
MWt	megawatt thermal
NASA	National Aeronautics and Space Administration
NBR	nucleate boiling ratio
NESC	National Energy Software Center
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
ORNL	Oak Ridge National Laboratory
OS	operating system
OT	operational transient
PCM	power-coolant mismatch
PCP	primary coolant pumps
PCS	primary coolant system
PNL	Pacific Northwest Laboratory
PPS	plant protection system
PSAR	preliminary safety analysis report
PWR	pressurized water reactor
PZR	pressurizer
RHR	residual heat removal
RIA	reactivity initiated accident
RPI	Rensselaer Polytechnic Institute
RSIC	Radiation Shielding Information Center
SAR	safety analysis report
SG	steam generator
SGHWR	steam generating heavy-water reactor
SNL	Sandia National Laboratories
SRI	Stanford Research Institute
SRL	Savannah River Laboratory

Acronym	Definition
TH	thermal-hydraulic
TMI	Three Mile Island
TOP	transient over-power accident
TREAT	Transient Reactor Test facility
TVA	Tennessee Valley Authority
UKAEE	United Kingdom Atomic Energy Establishment
UNI	United Nuclear Industries
WG	West Germany
$\bar{W}$	Westinghouse Electric Corporation
WREM	water reactor evaluation model
1-D	one-dimensional
2-D	two-dimensional
3-D	three-dimensional
1-ϕ	one-phase fluid
2-ϕ	two-phase fluid

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4416	D. W. Sasser
4756	D. Engi (3)
5000	J. K. Galt
5600	D. B. Shuster, Attn: A. A. Lieber, M. M. Newsom, 5620 R. C. Maydew 5630
5640	G. J. Simmons, Attn: R. J. Thompson, 5641 L. F. Shampine, 5642
5642	B. L. Hulme
8214	M. A. Pound
3141	L. J. Erickson (5)
3151	W. L. Garner (3)

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Org.	Bldg.	Name	Rec'd by	Org.	Bldg.	Name	Rec'd by